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TRITIUM-ASSISTED FUSION BREEDERS

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Lawrence Livermore National Laboratory

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PREFACE

It is well known that D-D fusion has more neutrons available for breeding than D-T fusion. Therefore, the question arose, "Could a D-D-based fusion breeder outperform a D-T-based fusion breeder?"

This study answers that question: though the support ratio offered by D-D fusion breeders can be ~50% higher than the support ratio of D-T fusion breeders, their performance is fairly close when measured by total cost of the fissile product. For the D-D basis, we used the WILDCAT design; and for the D-T basis, we chose the STARFIRE design. Because the WILDCAT designers incorporated more advanced fusion technology (higher magnetic fields, higher plasma pressure divided by magnetic field pressure, and better plasma containment), the D-D fusion breeder appears to be a later application of fusion. We recommend the fusion breeder program concentrate on D-T fusion breeders until new advances are made that will significantly improve the performance of the D-D fusion oreeder over and above that of the D-T fusion breeder.

Ralph W. Moir
Fusion Breeder Program

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TRITIUM-ASSISTED FUSION BREEDERS

ABSTRACT

This report undertakes a preliminary assessment of the prospects of tritium-assisted D-D fuel cycle fusion breeders. Two well documented fusion power reactor designs--the STARFIRE (D-T fuel cycle) and the WILDCAT (Cat-D fuel cycle) tokamaks--are converted into fusion breeders by replacing the fusion electric blankets with ^{233}U producing fission suppressed blankets; changing the Cat-D fuel cycle mode of operation by one of the several tritium-assisted D-D-based modes of operation considered; adjusting the reactor power level; and modifying the resulting plant cost to account for the design changes. Three sources of tritium are considered for assisting the D-D fuel cycle: tritium produced in the blankets from lithium or from ^{3}He and tritium produced in the client fission reactors.

The D-D-based fusion breeders using tritium assistance are found to be the most promising economically, especially the Tritium Catalyzed Deuterium mode of operation in which the ^{3}He exhausted from the plasma is converted, by neutron capture in the blanket, into tritium which is in turn fed back to the plasma. The number of fission reactors of equal thermal power supported by Tritium Catalyzed Deuterium fusion breeders is about 50% higher than that of D-T fusion breeders, and the profitability is found to be slightly lower than that of the D-T fusion breeders. Design and operating alterations might improve the economics of tritium-assisted fusion breeders to match and perhaps surpass that of D-T fusion breeders. As suggested by previous breeder studies, the profitability of the fusion-breeder versions of both STARFIRE and WILDCAT appears to be significantly better than the respective fusion-electric versions of these concepts.

1.0. INTRODUCTION

1.1. THE INTEREST IN FUSION BREEDERS

There are mounting indications[1-4] that the earliest significant contribution fusion can make to the energy economy is via fissile fuel producing fusion-fission hybrid reactors. These are also referred to as hybrid fuel factories or as Fusion Breeders (FB). The crucial question concerning the commercial potential of fusion breeders is their economics. By economics, we mean whether or not the cost of electricity (COE) generated in a nuclear energy economy based on FB-light water reactor (LWR) symbiosis (assumed to be the most common of the fission reactors) could be comparable to or lower than the COE for alternate nuclear energy systems (e.g., LWRs supported by enrichment facilities with the price of uranium ore set at the cost expected at the time of FB commercialization).

Recent FB studies[5] indicate that the FB-LWR symbiosis might compete economically with the conventional system of enrichment facilities--LWR (with reprocessing), when the year zero equivalent U_3O_8 cost exceeds $171/kg. The present spot price of yellow-cake is as low as $44/kg,[6] although a more representative value for the near future is $100/kg.

1.2. PURPOSE OF WORK

D-D-based fusion fuel cycles, in particular, the partially catalyzed deuterium (PCD) mode of operation[7] and tritium-assistance,[7] were recently proposed to offer higher than D-T fissile fuel production ability.[8] The purpose of this work is to assess the promise of the PCD and tritium-assisted modes of operation for FB applications, and to identify directions and questions deserving further investigation.

1.3. EXPLANATION OF PARTIALLY CATALYZED DEUTERIUM (PCD) AND TRITIUM ASSISTANCE

By PCD we mean the semicatalyzed deuterium (SCD) and tritium-catalyzed deuterium (TCD) fusion fuel cycles. The SCD is similar to the catalyzed deuterium (Cat-D) fuel cycle with the exception that as much of the ^3He

[produced in the $D(D,n)^3He$ reaction] as possible is extracted from the plasma. This is accomplished simply by not recirculating the ^3He, which has a low probability of fusing before leaking out from the plasma for the first time.[9] The TCD mode is similar to the SCD mode, except that the ^3He extracted from the plasma exhaust is transmuted into tritium by neutron capture in the blanket. This tritium is fed back to the plasma.[10]

Table 1 compares the energy and neutron balance of the PCD and conventional fuel cycles. The characteristics of the PCD fuel cycles pertain to ideal cycles in which none of the ^3He fuses and, in the case of TCD, all of it is fed back as tritium. The figure-of-merit used to judge the fuel production ability of the fusion fuel cycles is F/W--the net number of fissile fuel atoms produced per total nuclear energy generated in the reactor. The F/W values given in the table pertain to idealized leakage, structure and coolant-free beryllium-thorium fission-suppressed blankets. The values are from Table 28 of Ref. 4. We observe that the PCD fuel cycles offer a higher fuel production ability than both the D-T and Cat-D fuel cycles.

By tritium assistance we imply one of the following scenarios. In the case of the D-T fuel cycle, tritium assistance involves supplying part of the tritium needs of the fusion reactor from sources external to the reactor. This relieves the tritium breeding design goal for the blanket and, consequently, improves its fissile fuel production ability. The specific external source of tritium that we considered is the client fission reactor, which uses the fissile fuel supply provided by the fusion breeder. The idea is[11] to use neutrons that would be otherwise lost in the control of LWRs for tritium production. This can be done by using ^6Li (or ^3He) instead of ^{10}B as the control material.

In the case of the non D-T fuel cycles, tritium assistance implies either the use of tritium produced in the client fission reactors or tritium produced in the blanket of the fusion breeder. By not having to breed tritium (i.e., to produce at least one triton per fusion neutron), the design of blankets for tritium-assisted PCD hybrid reactors can be significantly simpler and perhaps safer than the design of blankets for both D-T fusion and hybrid reactors without tritium assistance.

Table 1. Neutron and energy balance of prime fusion fuel cycles.

Fusion Fuel-cycle		Number of Neutrons[a] (Energy)	Fusion energy[a]		
			Total (MeV)	Fraction in neutrons	F/W[b] (atom/MeV)
D-T (conventional)					
$D + T \rightarrow n + \alpha$		1 (14.07)	17.59	0.80	0.067
D-based (alternate)					
D-D	$D+D \rightarrow n+{}^3He$ $D+D \rightarrow p + T$	1/2 (2.45)	3.65	0.34	
Cat-D	D-D with products T & ^3He fusing.	1/2 (14.07) + 1/2 (2.45)	21.62	0.38	0.059
SCD	D-D with T fusion; ^3He extracted.	1/2 (14.07) + 1/2 (2.45)	12.44	0.66	0.085
TCD	D-D with T fusion; ^3He extracted, converted into T and fed back.	1 (14.07) + 1/2 (2.45) -1/2 (thermal)	21.24	0.72	0.080

[a] Normalized per one initiating fusion reaction. Energy is given in MeV. Equal number of $(D\text{-}D)_n$ and $(D\text{-}D)_p$ reactions is assumed. No ^3He is used in the SCD and TCD modes of operation.

[b] F/W ≡ Net number of fissile atoms produced/total amount of nuclear energy generated.

1.4. APPROACH AND SCOPE

Compared with D-T fusion breeders, PCD fusion drivers are expected to improve the fissile fuel production ability of the blanket on the one hand but to impose more severe plasma confinement requirements and to lead to lower fusion power density on the other hand. Consequently, the assessment of PCD fuel cycles requires an economic analysis that properly weights the improvement in the blanket performance against the penalty of more demanding confinement and lower power density. There are three major parts to the work--plasma performance analysis (Section 2), blanket performance analysis (Section 3), and economic analysis (Section 4). Additional considerations, including the feasibility and consequences of tritium assistance from client fission reactors, are also examined (Section 5).

Because this is a short preliminary feasibility study, our investigation is restricted to one of many possible blanket designs, and we use many simplifying assumptions and approximate calculational models.

Critical characteristics of PCD and tritium-assisted plasmas are determined with the aid of a simple machine-independent zero dimensional model. The results thus obtained are used only in a relative sense for identifying the promising fusion driver modes of operation and, in the case of the economic analysis, for extrapolating the performance of fusion drivers designed to operate on the Cat-D fuel cycle, to PCD fuel cycles.

We used a simplified blanket model of the fission suppressed type. The blanket is modeled after a single-zone gas-cooled, fission-suppressed blanket benchmark used in the Lawrence Livermore National Laboratory (LLNL) Fusion Breeder Program.[5] The concentration of the primary blanket constituents--ThO_2, beryllium, and Li_2O offering the highest F/W value for each type of fusion neutron source is determined using the one-dimension neutron transport code ANISN. The blanket geometry is fixed.

The commercial prospects of tritium-assisted and/or PCD fusion breeders are assessed by comparing the economics of several D-D-based and tritium-assisted fusion breeders with that of client fission reactors supported by fuel enrichment facilities, as well as with that of conventional D-T fusion breeders. Self-consistent designs of a D-T and Cat-D fusion power reactor provide the basis for the economic analysis. These are the

STARFIRE[12] and the WILDCAT[13] tokamaks. The WILDCAT design based on the D-D fuel cycle has a more demanding performance than the D-T fuel cycle STARFIRE; however, the two designs are used for comparison. The capital costs of the FBs are estimated by scaling the cost for pertinent components of the reference STARFIRE and WILDCAT fusion power reactors to account for the difference in the blanket composition, fusion and blanket power levels, and power recirculation requirements.

The economic prospects of the fusion breeders are assessed by considering two figures-of-merit:

1. The ratio of the net annual income from the sale of electricity and fissile fuel to the total annual expenses (including the actual operation and maintenance costs as well as the return on the capital investment). This figure-of-merit is a measure of the profitability of the FBs.
2. The levelized cost of the FB-produced fissile fuel. This is the price the FB operators would have to get for the fissile fuel in order to cover all of its expenses.

The FB bus-bar cost of electricity is assumed to be that of the conventional LWR. Similarly, the value assigned to the fissile fuel (for the first figure-of-merit) represents the cost that the LWRs could pay the FBs if their cost of electricity is to be the same as when supported by enrichment facilities.

The potential for tritium assistance from the client fission reactors is assessed by estimating the fraction of the fission born neutrons that are being absorbed in the control systems of typical LWRs. No specific modifications are being examined in the control systems of LWRs for using the otherwise wasted neutrons for tritium production. Consequently, the assessment of the promise of tritium assistance from client fission reactors is restricted to possible improvements in the performance of FBs.

We could not evaluate all potential contributions of the D-D-based tritium-assisted FBs quantitatively. One example is the relatively high support ratio offered by certain tritium-assisted FBs, which could allow these FBs to be concentrated in a relatively small number of sites. Another example is the source of ^{3}He obtained as a by-product of certain FBs (i.e., the SCD and SCD-T), which could provide the fuel basis for D-^{3}He fusion power reactors.

Even though the results are not based on detailed conceptual designs of FBs, they are believed to be rather generic, and the comparisons between the performance of the different FBs are expected to be consistent enough to provide reliable indications for the relative merits of the systems examined.

2.0. PROPERTIES OF PARTIALLY CATALYZED DEUTERIUM AND TRITIUM-ASSISTED PLASMAS

Properties of PCD and tritium-assisted plasmas were recently studied parametrically using a simple machine-independent model. These properties and their sensitivities to various independent variables and assumptions are summarized in Ref. 7.

For the purpose of this study the properties of selected PCD and tritium-assisted plasmas are calculated for confinement conditions similar to those of the WILDCAT tokamak.[13] These calculations are performed with the zero-dimensional multispecies model for particle conservation and energy balance described in Ref. 7, using the following assumptions: equal electron and ion temperatures and a cyclotron radiation loss-rate coefficient[9] of Cs = 0.0121. This coefficient accounts for the effect of ducts, wall reflectivity, and plasma β. (The resulting cyclotron losses calculated for a Cat-D plasma being at the WILDCAT conditions agree with the WILDCAT cyclotron losses.)[13] We also assumed that the ^{3}He concentration, and therefore burn in the plasma, is kept at its lowest possible value (i.e., there is no recirculation of ^{3}He); the confinement time of all ash and fuel species is identical and equal to the energy confinement time; and finally that the total plasma pressure is the same for all plasmas.

Table 2 compares our simple model prediction with the WILDCAT design[13] characteristics of the Cat-D plasma. The plasma temperature for the simple model calculation is selected so that the calculated fusion energy gain will assume the WILDCAT design value of about 20. We observe that, relative to the WILDCAT design model,[13] the simple model underestimates the power loss by leakage but overestimates the radiation power losses. The simple model predictions could have been adjusted to better match the WILDCAT characteristics by using the energy and particle confinement laws of Ref. 13.

Table 2. Comparison of our simple model predictions and the design characteristics of the Cat-D WILDCAT plasma.

Parameter	Design (Ref. 13)	Simple model[a]
Electron temperature (keV)	30	37.5[b]
Ion temperature (keV)	32 to 52[c]	37.5[b]
Fusion energy gain (Q)	20	20
Relative ion density		
D	0.8220	0.8360
T	0.0040	0.0037
^3He	0.0919	0.123
α	0.0242	0.018
P	0.0580	0.018
Relative power loss		
Leakage	0.347	0.235
Bremsstrahlung	0.405	0.674
Cyclotron	0.069	0.092

[a] Imposing a confinement parameter of $n_e\tau = 4.1 \times 10^{21}\ m^{-3}\ s$.

[b] The plasma temperature is imposed so that Q will assume the WILDCAT design value of $\sim$20 (under the assumption of $T_e = T_i$ and other assumptions described in this section).

[c] This range reflects the actual plasma temperature profile. The electron temperature given is the average temperature.

We expect, nevertheless, that the relative behavior of the different D-D-based plasmas will be predicted by the simple model with sufficient accuracy for the purposes of this study.

Figures 1 through 3 show properties of certain PCD and tritium-assisted plasmas under the confinement ability of the WILDCAT design,[13] which is $n_e\tau \sim 4 \times 10^{21}\ m^{-3}$ s. The degree of tritium-assistance is denoted by γ_x, the number of tritons fed back to the plasma per fusion neutron of type x, which is either D-T or D-D. Shown also in the figures is the performance of the reference Cat-D plasma.

We see that the tritium-assisted PCD plasmas (i.e., TCD and SCD-T) offer a significantly higher D-T and D-D neutron yield densities, and (Fig. 3) higher fusion power density (Fig. 2) than the WILDCAT Cat-D plasma, when all plasmas are in the vicinity of the WILDCAT design temperature. The SCD mode of operation, on the other hand, exhibits almost no gain in the fusion neutron yield, some loss in the fusion power density (Fig. 2), and a significantly lower fusion energy gain (Fig. 1).

There are a number of ways to increase the PCD plasmas fusion energy gain, including:

(1) Increasing the plasma temperature, as illustrated in Fig. 1.

(2) Assisting the plasma with tritium. Illustrated in Fig. 1 are two modes of tritium assistance--the TCD mode for which $\gamma_{DT} = 0.4$ and $\gamma_{DD} = 0.2$, and the SCD-T mode with $\gamma_{DT} = \gamma_{DD} = 0.5$. With the latter degree of tritium assistance, the SCD-T fusion energy gain almost matches that of the reference Cat-D. Additional illustration of the effect of tritium-assistance is given in Fig. 4.

(3) Increasing the fraction of the ^{3}He, which fuses in the plasma (Fig. 5).

(4) Increasing the plasma confinement parameter, $n_e\tau$ (Fig. 6), an approach that is beyond the scope of this work. The necessary and most practical approach for fusion energy gain enhancement is a matter for an overall system economic analysis.

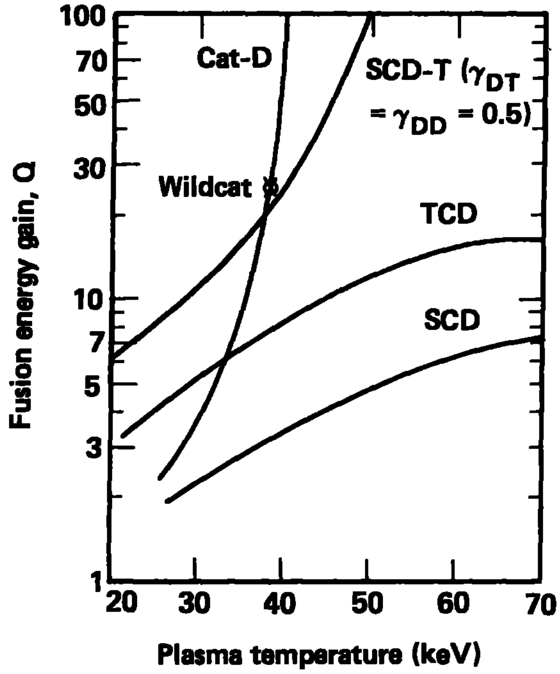

Figure 1. Fusion energy gain attainable from selected PCD plasmas and Cat-D plasma. $n_e\tau = 4 \times 10^{21}\ m^{-3}s$.

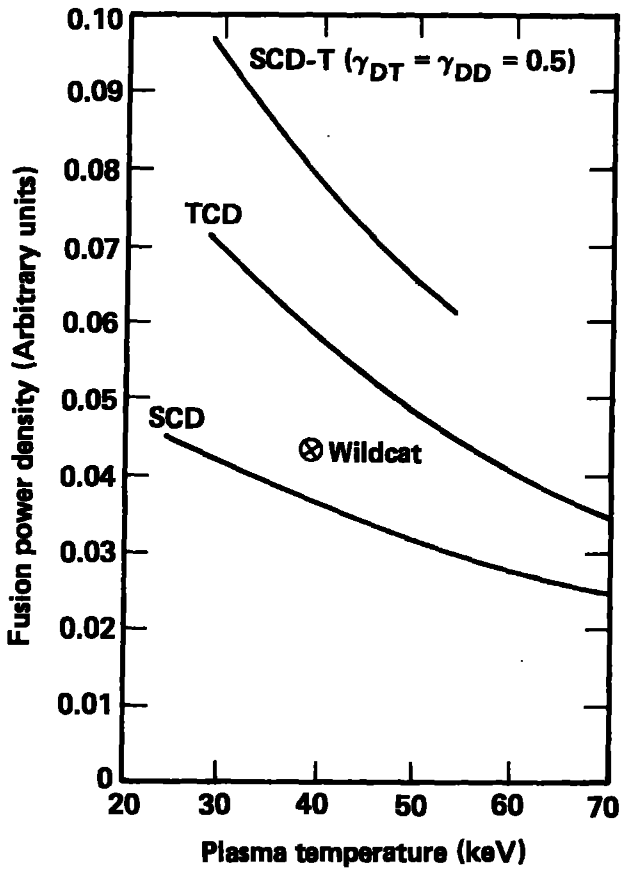

Figure 2. Fusion power density of selected PCD plasma compared with the WILDCAT Cat-D plasma. $n_e\tau = 4 \times 10^{21}\ m^{-3}s$.

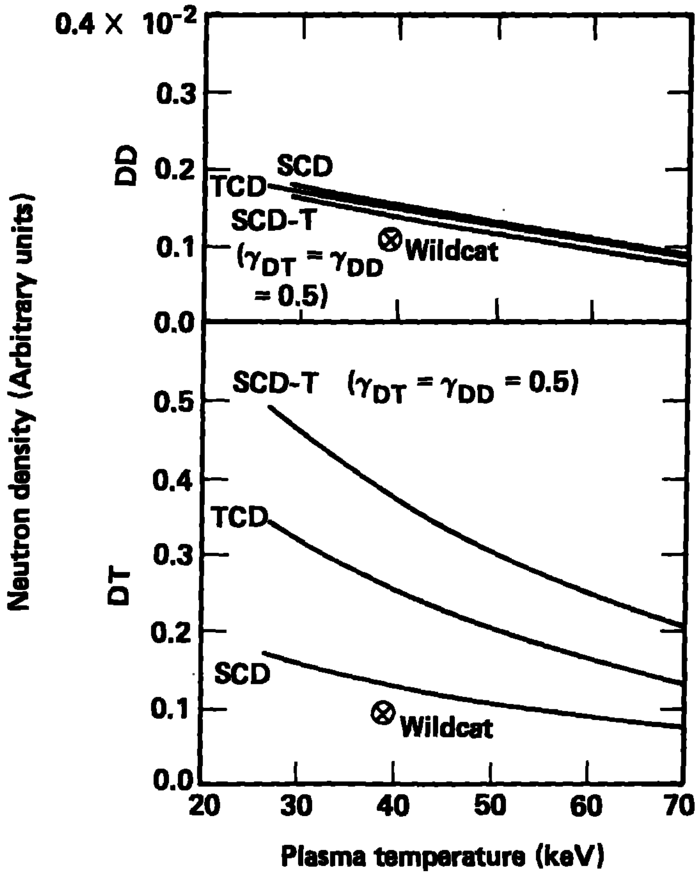

Figure 3. Fusion neutron density of selected PCD plasmas compared with the WILDCAT Cat-D plasma. $n_e\tau = 4 \times 10^{21}$ $m^{-3}s$.

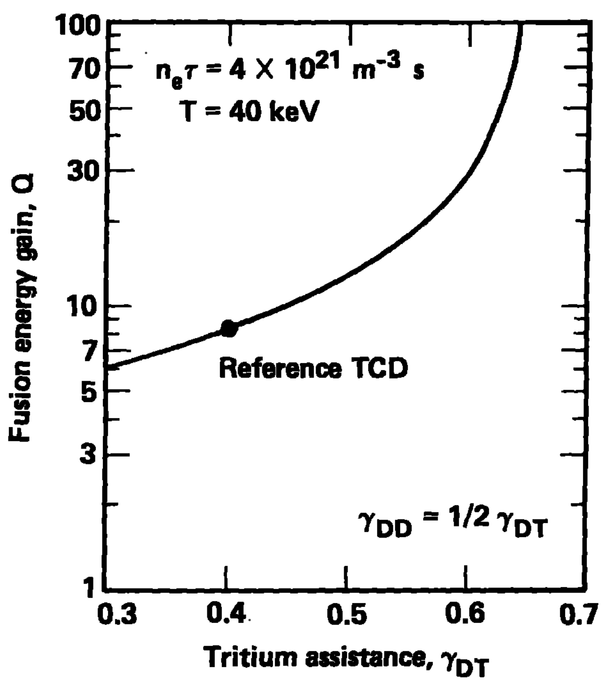

Figure 4. Effect of degree of tritium (γ_{D-T}) on the fusion energy gain of PCD plasmas.

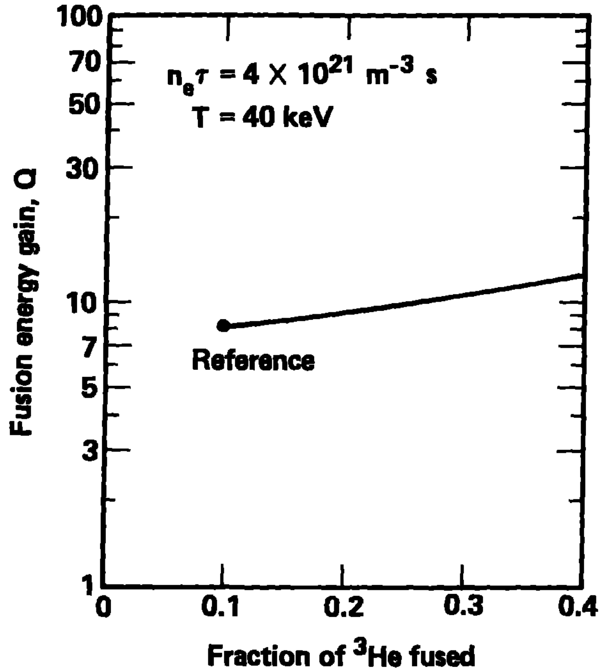

Figure 5. Effect of the fraction of ^3He fusing, on the fusion energy gain of TCD plasmas.

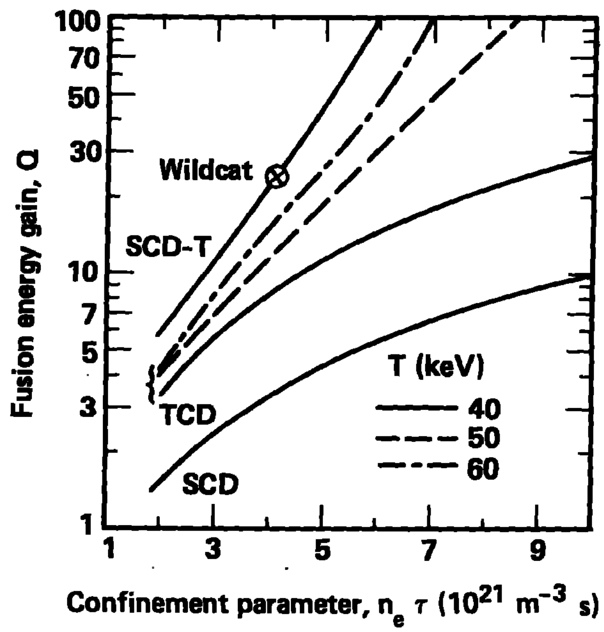

Figure 6. Effect of the confinement parameter on the fusion energy gain of selected plasmas.

3.0. PROPERTIES OF BLANKETS

3.1. BLANKET DESCRIPTION

For this study, we selected the beryllium blanket benchmark used in the fusion breeder program[5] as the reference blanket for comparing nuclear data and calculational methods pertaining to the gas cooled blanket design. This blanket does not correspond to a final design concept, but is representative of fission-suppressed blankets featuring beryllium neutron multiplication and gas cooling.[5]

Figure 7 shows the geometry and composition of the reference blanket. To further simplify the analysis, we assumed a slab geometry preserving the zone thicknesses of Fig. 7. The blanket design variables that we considered are the concentration of the ThO_2, beryllium, and Li_2O (the lithium is of natural isotopic composition), as well as the concentration of ^{233}U. No fission products and higher actinides are accounted for here. The iron volume fraction is constant and so is the coolant volume fraction (the coolant is not represented explicitly, as it does not affect the blanket neutronics). In blankets to be driven by TCD fusion devices, ^3He is assumed to replace Li_2O for the production of tritium. The ^3He is either homogeneously distributed throughout the blanket, possibly serving as the blanket coolant, or it is contained in a thin layer adjacent to the first wall, possibly serving as the first wall coolant. The primary composition variable is ThO_2. The concentration of Li_2O (or ^3He) is sought to provide the desired tritium production, so that the adjustable variable is the beryllium volume fraction.

Even though very simplified, this blanket model should properly bear out the relative performance of gas-cooled beryllium fission-suppressed blankets driven by the various fusion neutron sources considered. Moreover, this relative performance is not expected to be very sensitive to the chemical form of the blanket constituents.

3.2. CALCULATIONAL MODEL

All the neutronic calculations were performed with the one-dimensional transport code ANISN[14], using the S_4-P_1 transport approximation, with an

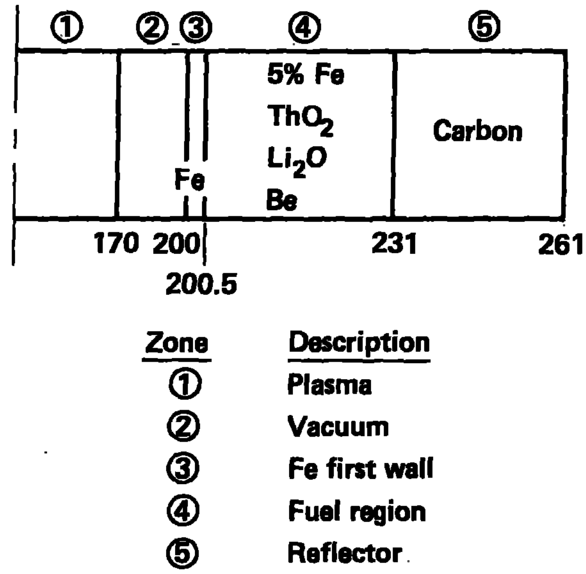

Figure 7. Our gas-cooled beryllium fission-suppressed blanket model. The concentration of zone 4 constituents is variable.

albedo of 0.3 assigned to the right boundary. A 15-neutron, 7-photon, coupled cross section set was used for the calculations. The 22 group constants were derived[15] using a 1/E weighting for the neutrons and constant weighting for the photons from the 100n-21γ EPR library.[16,17] Kerma factors and cross sections for specific reactions were derived[15] from the MACLIB-IV library[18,19] using the MAC-IV program.[20] These cross sections pertain to infinite dilution and 800 K. The energy group structure, taken from Ref. 21, is shown in Table 3. Additional technical details on the preparation of the group constants are given in Appendix A.

We believe the accuracy of the low-order transport approximation and the coarse group structure used here are adequate for our present purposes. This is illustrated in Table 4, which compares performance parameters of the reference blanket (Fig. 7) based on the model and data described above with the corresponding parameters calculated with the standard tools and nuclear data used in the fusion breeder program.[5] The agreement between our simple model and the results calculated at LLNL[5] is generally satisfactory.

Table 3. Coupled 15-neutron, 7-gamma group structure.

Group number	Neutron upper energy (eV)	Group number	Gamma upper energy (eV)
1	1.4918 + 7	16	1.4 + 7
2	1.3499 + 7	17	8.0 + 6
3	1.2214 + 7	18	6.5 + 6
4	1.1052 + 7	19	5.0 + 6
5	6.0653 + 6	20	3.5 + 6
6	3.3287 + 6	21	2.0 + 6
7	2.4660 + 6	22	4.0 + 5
8	2.2313 + 6		
9	1.0026 + 6		
10	2.4724 + 5		
11	3.1828 + 4		
12	3.5358 + 2		
13	2.9023 + 1		
14	1.4450		
15	0.4140		

Table 4. Comparison of the simple model results for the beryllium blanket benchmark with results calculated using the ANISN/ENDF and TARTNP/ENDL systems. Normalized per one 14-MeV neutron.

Performance parameter	TARTNP/ENDL		ANISN/ENDF			This study	
	0% ^{233}U	0.25% ^{233}U	0% ^{233}U	0.1% ^{233}U	2% ^{233}U	0% ^{233}U	0.1% ^{233}U
Tritons produced (T)	0.981	0.996	1.01	1.03	1.18	0.990[a]	0.996[a]
$^{6}Li(n,t)$	0.977	0.991				0.958	0.964
$^{7}Li(n,n't)$	0.004	0.005				0.005	0.005
Net fissile bred (F)	0.86	0.85	0.93	0.92	0.86	0.86	0.85
Total fission rate	0.007	0.038	0.0068	0.023	0.38	0.0066	0.019
Be(n,2n)	1.32	1.32	1.39	1.39	1.50	1.36	1.37
Leakage	0.08	0.09				0.054	0.056
Blanket energy multiplication[b]	1.51	1.93	1.6	1.8	6.6	1.65	1.82
T + F	1.84	1.845	1.94	1.95	2.04	1.848[a]	1.848[a]

[a] Including 0.0265 tritons produced in the beryllium.

[b] Total energy (MeV) deposited in the blanket per fusion neutron/14.1.

3.3. FISSILE FUEL PRODUCTION ABILITY

3.3.1. Tritium Breeding D-T Driven Blankets

Table 5 and Fig. 8 show the effect of the ThO_2 volume fraction, and the correspondingly adjusted Li_2O and beryllium volume fractions, on the properties of D-T driven blankets designed to give approximately 1.1 tritons (gross) per D-T neutron. We assume ^{233}U to be present in these blankets with an average concentration of 0.1% (atomic percent) of the thorium.

The highest fuel production ability, measured in terms of both the absolute production per fusion neutron (F + T) and in terms of the production per unit nuclear energy deposited and generated by the fusion neutrons in the blanket, which is proportional to (F + T)/M, is obtained with a ThO_2 volume fraction of about 8%. At higher ThO_2 concentrations, the beryllium volume fraction starts to decline rapidly as the volume fraction of Li_2O quickly increases. Being the primary neutron multiplier, the beryllium content decline reduces the blanket fuel production ability. Below the optimal ThO_2 concentration, a reduction in the ThO_2 (and therefore also in the Li_2O) volume fraction leads to an enhancement in the parasitic neutron capture.

The Th/Be atom density ratio offering the highest F/M, i.e., fissile fuel production ability, is about 50% higher than the corresponding ratio used for the reference blanket of the 1982 FB program.[5] Had the present blanket been designed to have 2/3 of the optimal Th/Be atomic ratio, i.e., with a reduced ThO_2 volume fraction, its F/M would have been about 95% of the maximum attainable (see Fig. 2).

Based on this analysis, we expected that enriching the Li_2O with ^6Li is likely to improve its fuel production ability, with the optimal concentration of ThO_2 shifting towards higher volume fractions.

3.3.2. Tritium-Breeding-Free D-T-Driven Blankets

Tables 6 and 7 and Fig. 9 summarize the fuel production ability of D-T driven blankets that are free from the need to produce any tritium, i.e., free from Li_2O, and that contain either 0% or 0.1% ^{233}U.

Table 5. Performance parameters of tritium breeding D-T driven blankets. Average ^{233}U concentration is 0.1%.

Parameter	ThO$_2$ volume fraction (%) 4	6	12
Li_2O volume fraction (%)	1.5	4.0	20.0
Tritium production (per D-T n)			
^{6}Li	1.0865	1.0858	0.9439
^{7}Li	0.0076	0.0199	0.0933
Be	0.0264	0.0250	0.0184
Total (T)	1.1205	1.1307	1.0556
Fissile fuel production (per D-T n)			
Gross	0.8229	0.9343	0.8803
Net (F)	0.8090	0.9244	0.8751
Blanket energy (MeV)	25.13	24.72	24.02
Net fissile fuel production + tritium production (F + T)	1.9295	2.0551	1.9307
(F + T)/M[a]	1.0827	1.1720	1.1334
Absorptions (per D-T n)			
First wall	0.0989	0.0609	0.0362
Blanket Fe	0.1697	0.0783	0.0322
Be	0.1827	0.1625	0.1106
^{6}Li	1.1037	1.0906	0.9425
Th	0.8710	0.9667	0.9194
^{233}U	0.0139	0.0099	0.0052
Leakage	0.0524	0.0437	0.0336
Fissions/ν x Fissions			
Th	0.0066/0.0223	0.0097/0.0330	0.0185/0.0630
^{233}U	0.0103/0.0309	0.0079/0.0216	0.0045/0.0114
Be(n,2n)	1.3587	1.2672	0.8780
Th(n,2n) + Th(n,3n)	0.0258	0.0382	0.0736

[a]M = Blanket energy (MeV)/14.1.

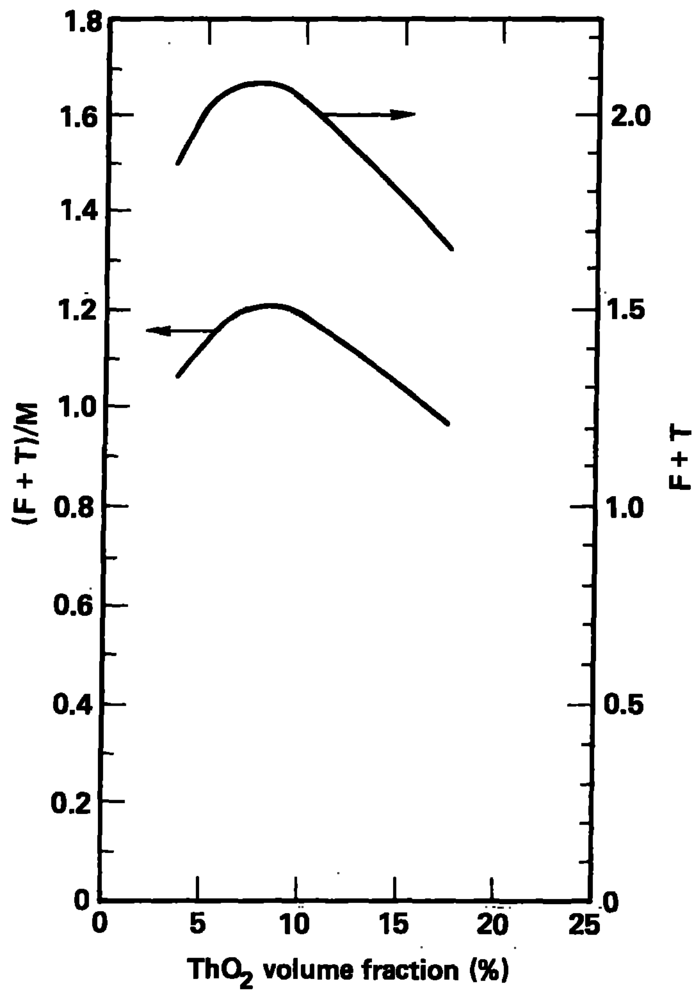

Figure 8. Fuel production ability of tritium breeding D-T-driven blankets. ^{233}U concentration is 0.1%.

Table 6. Performance parameters of tritium breeding-free D-T driven blankets without ^{233}U.

Parameter	ThO_2 volume fraction (%)				
	4	8	16	35	95
Fissile fuel production					
Net (F)	1.1221	1.5395	1.8217	1.8690	1.6513
Blanket energy (MeV)	24.35	24.28	25.28	27.99	34.34
F/M[a]	0.6497	0.8939	1.0160	0.9414	0.6780
Beryllium tritium production	0.0265	0.0253	0.0230	0.0174	0
Absorptions					
First wall	0.2318	0.1378	0.0702	0.0377	0.027
Blanket Fe	0.6080	0.3089	0.1170	0.0378	
Be	0.2256	0.1837	0.1449	0.0961	
Th	1.3077	1.7243	1.9566	1.9590	
Leakage	0.0846	0.0598	0.0430	0.0311	0.039
Fissions	0.0066	0.0129	0.0249	0.0494	0.1053
ν x Fissions	0.0223	0.0437	0.0848	0.1700	0.3685
Be(n,2n)	1.3641	1.2691	1.0967	0.7396	0
Th(n,2n) + Th(n,3n)			0.0986	0.1987	0.4204

[a] M = Blanket energy (MeV)/14.1.

Table 7. Performance parameters of tritium breeding-free D-T driven blankets with 0.1% ^{233}U.

Parameter	ThO_2 volume fraction (%)				
	4	8	16	35	95
Fissile fuel production					
Gross	1.1643	1.6021	1.8691	1.8922	1.6695
Net (F)	1.1179	1.5554	1.8360	1.8756	1.6555
Blanket energy (MeV)	30.26	30.32	29.79	30.52	36.73
F/M[a]	0.5210	0.7234	0.8691	0.8664	0.6355
Beryllium tritium production	0.0265	0.0254	0.0230	0.1736	0
Absorptions					
First wall	0.2346	0.1383	0.0698		0.0266
Blanket Fe	0.6224	0.3122	0.1159		0.0185
Be	0.2317	0.1898	0.1479		0
Th	1.3531	1.7884	2.0021		1.7958
^{233}U	0.0464	0.0467	0.0331	0.0166	0.0140
Leakage	0.0929	0.0660		0.0324	0.0406
Fissions/ν x Fissions					
Th	0.0067/0.0225	0.0131/0.0444	0.0251/0.0854	0.0497/0.1706	0.1060/0.3699
^{233}U	0.0302/0.1051	0.0313/0.1053	0.0235/0.0742	0.0133/0.0374	0.0123/0.0330
Be(n,2n)	1.3777	1.2896	1.1051	0.7427	0
Th(n,2n) + Th(n,3n)	0.0258	0.0510	0.0987	0.1988	0.4203

[a]M = Blanket energy (MeV)/14.1.

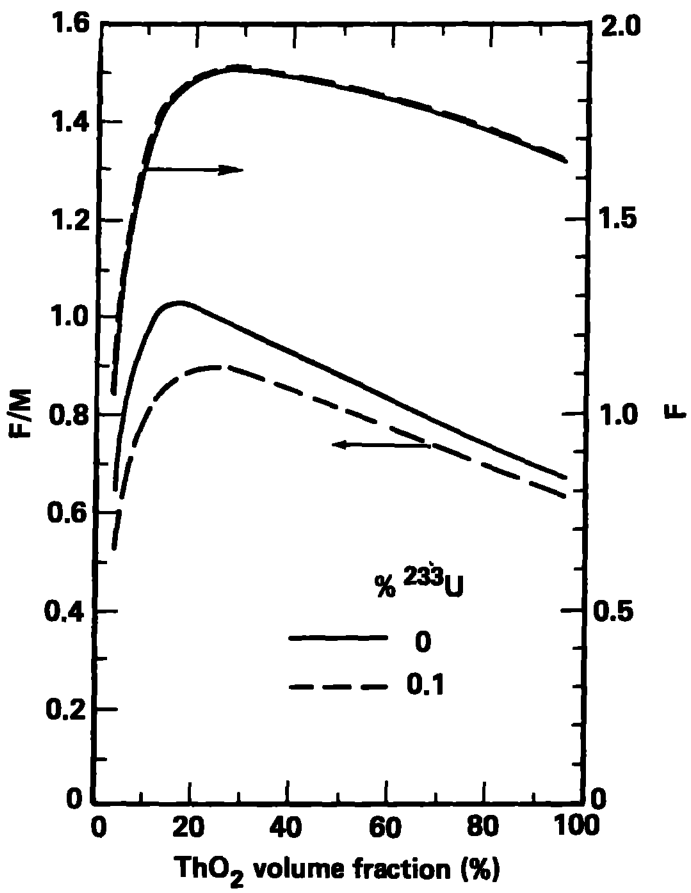

Figure 9. Fuel production ability of tritium breeding-free D-T-driven blankets.

Consider, first, the 0.1% ^{233}U case. The maximal number of fissile atoms (net) produced per D-T neutron (F) is about 1.88, attained with a ThO_2 volume fraction of about 25% (Fig. 9). This is to be compared with a maximal F + T of about 2.08 (implying a F < 1.08) offered by the tritium breeding blanket having a ThO_2 volume fraction of 7.5% (Fig. 8).

As measured by the sum of ^{233}U and tritium production, the lower neutron utilization of the lithium-free blanket results from an increase in the parasitic neutron capture. Compare the number of neutrons absorbed in the first wall and blanket iron structure (Tables 5 and 7). The shift of the optimal ThO_2 volume fraction to a higher value in the no-lithium case is caused by the higher beryllium content in the no-lithium blankets.

Comparing the fissile fuel production ability of blankets with and without tritium breeding, we find that

$$F(T = 0)/F(T = 1.0) = 1.74 \ ; \qquad F(T = 0)/F(T = 1.1) = 1.92 \ ;$$

$$\frac{F/M(T = 0)}{F/M(T = 1.0)} = 1.44 \ ; \quad \frac{F/M(T = 0)}{F/M(T = 1.1)} = 1.59 \ .$$

We see that whereas the gain in the number of fissile atoms produced because of the elimination of the tritium breeding requirement is in the vicinity of 80%, the gain in F/M is only of the order of 50%. The lower gain in F/M is attributed to the higher energy multiplication of the lithium-free blankets (compare Tables 5 and 7). This reduction in gain is caused, primarily, by enhanced fissions of ^{233}U.

Indeed, the F/M value of the lithium-free blanket having no ^{233}U can reach 1.03 vs 0.90 in the presence of 0.1% ^{233}U (see Fig. 9). The maximal value of F/M with no ^{233}U is obtained at a lower ThO_2 volume fraction (near 15%). This occurs because the contribution of ^{233}U to the fissions in the blanket decline with the increase in the thorium and ^{233}U content (see Table 7), i.e., with the increased probability for epithermal neutron capture in the thorium resonances.

It is interesting to note that the ^{233}U has a negligible effect on F; the loss of ^{233}U just about compensates for the increase in the number of ^{233}U atoms produced (see Fig. 9 and Tables 6 and 7).

3.3.3. Tritium-Breeding-Free D-D-Driven Blankets

Tables 8 and 9 and Fig. 10 show the fuel production ability of D-D-driven blankets that are free of lithium. Even though the number of fissile atoms producible by a D-D neutron is significantly smaller than that producible by a D-T neutron by about a factor of two, the D-D F/M value is significantly higher. This is because of the lower kinetic energy of the D-D fusion neutron, which is deposited in the blanket, and because of the lower probability of this neutron to induce fissions, primarily in thorium.

Whereas the shape of the F/M vs ThO_2 volume fraction for D-D neutrons is similar to that for D-T neutrons (compare Figs. 9 and 10, both with and without 0.1% ^{233}U), the shape of the F vs ThO_2 volume fraction is different--increasing monotonically with the contents of thorium and reaching its maximum for an all-ThO_2 blanket. This monotonous increase with the thorium content reflects the fact that the 2.45-MeV neutron has a very low probability for inducing n,2n reactions in the beryllium. In other words, beryllium serves no useful purpose in the case of the D-D neutrons, whereas the increase in the thorium content reduces the parasitic neutron captures, primarily in iron, thus increasing F (see Table 9).

The effects of ^{233}U in the D-D-driven blankets is similar to its effects in the D-T-driven blankets; F(net) is insensitive to an increase in the ^{233}U content, while F/M declines. F/M in the D-D-driven blankets is somewhat more sensitive to the ^{233}U content than in the D-T case, as the relative contribution of the ^{233}U fissions to the blanket energy production is higher for the D-D case.

3.3.4. Blankets for Tritium-Catalyzed Deuterium and Tritium-Assisted Drivers

The TCD drivers require the conversion of ^3He into tritium. The fraction of ^3He recoverable from typical TCD plasmas is on the order of 0.9.[9] The question is how much tritium the blanket must be designed to produce per D-T and D-D neutron to convert the ^3He into tritium at the rate of its production.

Assuming that the number of tritons (γ_{DT}) produced per D-T neutron is twice that produced per D-D neutron (γ_{DD}), an assumption that is based

Table 8. Performance parameters of tritium breeding-free D-D driven blankets without ^{233}U.

Parameter	ThO_2 volume fraction (%)				
	4	8	16	35	95
Fissile fuel production					
Net (F)	0.4761	0.6681	0.8231	0.9220	0.9962
Blanket energy (MeV)	7.8236	7.6534	7.7737	8.9485	13.4802
F/M[a]	0.8580	1.2309	1.4929	1.4528	1.0420
Absorptions					
First wall	0.1012	0.0578	0.0257	0.0098	0.0045
Blanket Fe	0.2499	0.1264	0.0467	0.0505	0.0067
Be	0.0898	0.0753	0.0636	0.0135	
Th	0.5516	0.7430	0.8733	0.9508	
Leakage	0.0176	0.0117	0.0073	0.0062	0.0224
Fissions	0.0011	0.0022	0.0044	0.0103	0.0375
ν x fissions	0.0023	0.0047	0.0097	0.0225	0.0824
Be(n,2n)	0.0093	0.0093	0.0093	0.0092	
Th(n,2n) + Th(n,3n)					0.0006

[a] M = Blanket energy (MeV)/14.1.

Table 9. Performance parameters of tritium breeding-free D-D driven blankets with 0.1% ^{233}U.

Parameter	ThO_2 volume fraction (%)				
	4	8	16	35	95
Fissile fuel production					
Gross	0.4940	0.6930	0.8442	0.9330	1.0069
Net (F)	0.4746	0.6732	0.8297	0.9252	0.9984
Blanket energy (MeV)	10.2539	10.8755	9.8969	10.1599	14.9269
F/M[a]	0.6526	0.8728	1.1821	1.2840	0.9431
Absorptions					
First wall	0.1023	0.0580	0.0255	0.0097	0.0045
Blanket Fe	0.2547	0.1275	0.0463	0.0134	0.0067
Be	0.0924	0.0776	0.0650	0.0511	
Th	0.5708	0.7682	0.8955	0.9614	1.0569
^{233}U	0.0194	0.0198	0.0145	0.0078	0.0085
Leakage	0.0202	0.0126	0.0081	0.0066	0.0233
Fissions/ν x Fissions					
Th	0.0011/0.0024	0.0022/0.0049	0.0046/0.0099	0.0105/0.0228	0.0379/0.0834
^{233}U	0.0126/0.0438	0.0133/0.0445	0.0104/0.0324	0.0063/0.0175	0.0075/0.0195
Be(n,2n)		0.0150	0.0132	0.0108	
Th(n,2n) + Th(n,3n)					0.0008

[a] M = Blanket energy (MeV)/14.1.

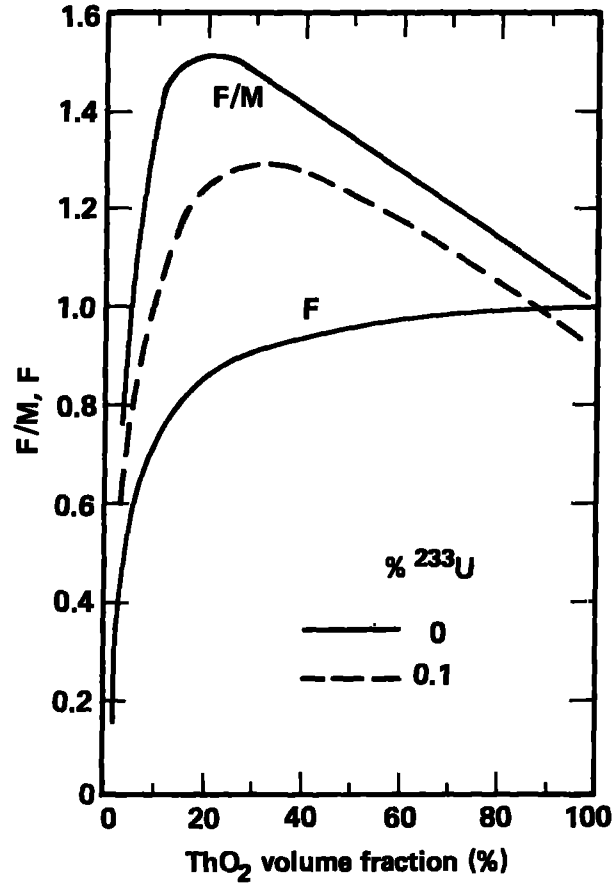

Figure 10. Fuel production ability of tritium breeding-free D-D-driven blankets.

on our experience and will be checked shortly by comparing the F values of Table 6 and 8, we find that the ^3He and tritium balance require

$$0.9 <\sigma_{DD}v>_{^3He} = <\sigma_{DD}v>_{^3He}\gamma_{DT}/2 + [<\sigma_{DD}v>_T + 0.9 <\sigma_{DD}v>_{^3He}]\gamma_{DT} \ .$$

In the above $<\sigma_{DD}v>_{^3He}$ and $<\sigma_{DD}v>_T$ are the fusion reactivities for, respectively, the $D(D,n)^3He$ and the D(D,p)T reactions. For a 40-keV plasma, this equation gives $\gamma_{DT} \sim 0.4$, so that $\gamma_{DD} \sim 0.2$.

Tables 10 and 11 summarize performance parameters of three blankets designed to convert the ^3He into tritium. In two of the designs, the ^3He is concentrated in a narrow zone between the first wall and the blanket (i.e., it is localized). For the numerical computations, the effective zone thickness is taken to be 1.5 cm, whereas the ^3He atomic density is adjusted to provide the desired transmutation rate. In another design the ^3He is distributed uniformly throughout the blanket, just as is lithium in the blankets for D-T drivers. This ^3He, possibly diluted with ^4He, could serve as the blanket coolant. Since we did not have the ^3He in our cross-section library, we inferred the performance of the TCD blankets from calculations of Li_2O containing blankets; consequently, we expect the results to underestimate the fuel production ability of TCD blankets.

A comparison of the neutron balance in the TCD and tritium-breeding-free blankets (Tables 10 and 11 vs Tables 8 and 9) shows that the addition of ^3He or ^6Li to the blanket, as required by the TCD or tritium-assisted modes of operation, significantly reduces the parasitic neutron captures, particularly in the first wall. This leads to an improvement in the overall neutron utilization as measured in terms of F + T. Moreover, the addition of ^3He significantly reduces the ^{233}U fission probability, thus reducing M. Also contributing to the lower M of the TCD blankets are the direct neutron captures in ^3He--the binding energy released in these captures (0.76 MeV) is significantly lower than that released when neutrons are captured in the blanket structure or in the fuel. Consequently, the TCD blanket F/M is within about 10% of the tritium-breeding-free F/M. (The TCD F/M is higher for the 8% and lower for the 16% ThO_2 volume fraction blankets.)

When distributed throughout the blanket, the total inventory of ^3He needed is smaller by a few folds than that needed in the localized arrangement. The distributed ^3He distribution is also more effective in

Table 10. Performance parameters of blankets for TCD and tritium-assisted fusion freeders with 0.1% ^{233}U, subjected to a D-T neutron.

Parameter	ThO_2 volume fraction (%)		
	8	16	12
^3He insertion	Localized[a]	Localized[a]	Distributed[b]
atomic density	7.0×10^{-4}	1.4×10^{-3}	1.1×10^{-5}
Tritium production			
^3He	0.4322	0.3568	0.3812
Be	0.0251	0.0224	0.0240
Total (T)	0.4573	0.3792	0.4052
Fissile fuel production			
Gross	1.3684	1.5762	1.5726
Net (F)	1.3328	1.5507	1.5503
Net fissile fuel production			
+ tritium production (F + T)	1.7901	1.9299	1.9555
Blanket energy (MeV)	26.6053	26.8484	26.2436
(F + T)/M[c]	0.9487	1.0135	1.0506
(F + T - 0.4)/M	0.7367	0.8035	0.8356
Absorptions (per D-T n)			
First wall	0.0383	0.0324	0.0670
Blanket Fe	0.2391	0.0918	0.0988
Be	0.1807	0.1432	0.1553
^3He	0.4322	0.3568	0.3812
Th	1.5106	1.6818	1.6555
^{233}U	0.0356	0.0255	0.0223
Leakage	0.0646	0.0451	0.0443
Fissions/ν x Fissions			
Th	0.0130/0.0439	0.0247/0.0838	0.0191/0.0647
^{233}U	0.0241/0.0802	0.0183/0.0571	0.0167/0.0496
Be(n,2n)	1.2765	1.0868	1.1832
Th(n,2n) + Th(n,3n)	0.0505	0.0970	0.0748

[a] The ^3He is localized in a 1.5 cm thick zone in between the first wall and the blanket.
[b] The ^3He is uniformly distributed across the 30.5 cm thick blanket.
[c] M = Blanket energy (MeV)/14.1.

Table 11. Performance parameters of blankets for TCD and tritium-assisted fusion breeders with 0.1% ^{233}U, subjected to a D-D neutron.

Parameter	ThO_2 volume fraction (%) 8	16	12
^3He insertion	Localized[a]	Localized[a]	Distributed[b]
atomic density	7.0×10^{-4}	1.4×10^{-3}	1.1×10^{-5}
Tritium production	0.2215	0.1976	0.1651
Fissile fuel production (T)			
Gross	0.5742	0.6887	0.6995
Net (F)	0.5599	0.6782	0.6898
Net fissile fuel production			
+ tritium production (F + T)	0.7814	0.8758	0.8549
Blanket energy (MeV)	8.3670	8.3906	8.4208
(F + T)/M[c]	1.3168	1.4717	1.4315
(F + T - 0.2)/M	0.9798	1.1356	1.0966
Absorptions (per D-D n)			
First wall	0.0092	0.0064	0.0239
Blanket Fe	0.0912	0.0338	0.0385
Be	0.0726	0.0618	0.0656
^3He	0.2215	0.1976	0.1651
Th	0.6276	0.7260	0.7305
^{233}U	0.01426	0.0105	0.0097
Leakage		0.0008	0.0068
Fissions/ν x Fissions			
Th	0.0022/0.0048	0.0044/0.0096	0.0033/0.0072
^{233}U	0.0097/0.0320	0.0076/0.0236	0.0073/0.0216
Be(n,2n)	0.0132	0.0118	0.0120

[a]The ^3He is localized in a 1.5 cm thick zone in between the first wall and blanket.

[b]The ^3He is uniformly distributed across the 30.5 cm thick blanket.

[c]M = Blanket energy (MeV)/14.1.

reducing the ^{233}U fission probability and blanket parasitic captures, but it is less effective in reducing the first wall captures.

Rather than using ^{3}He in the blanket, the same amount of tritium can be produced using ^{6}Li, possibly in the form of natural lithium, allowing us to attain a similar neutron balance. The energy multiplication of the lithium containing blankets will be somewhat higher, though, because of the larger binding energy released in the $^{6}Li(n,\alpha)T$ reaction (4.78 MeV). Moreover, the neutron multiplication and, therefore, (F + T) are expected to be somewhat reduced with the use of lithium. Such blankets are suitable for tritium-assisted SCD fusion breeders. The performance of such FBs will be similar to that of TCD FBs, except for a somewhat smaller support ratio. On the other hand, the tritium-assisted SCD reactors offer a prolific source of ^{3}He--possibly for future D-^{3}He fusion power reactors.

3.3.5. Comparison of Blankets

Tables 12 through 14 compare the fissile fuel production ability of three blankets that differ in their tritium production requirements, driven by four different fusion neutron sources. (The Cat-D and SCD drivers use the same type of blanket.) The performance of the various blankets is taken to be close to the optimal identified in the preceding subsections. The D-D-based drivers are assumed to operate with an average plasma temperature of 40 keV and, in the case of the SCD and TCD plasmas, to burn 10% of the ^{3}He they produce.

The three tables differ in the set of assumptions used concerning the blanket neutron utilization efficiency. Table 12 pertains to ideal systems in which all the fusion neutrons reach the blanket (i.e., the blanket coverage efficiency is 100%) and there is no loss of tritium and fissile fuel. Table 13 pertains to blankets with neutron efficiency of 95% and 85% respectively, for the production of tritium and fissile fuel. The fuel production ability of these blankets is deduced from the results obtained for the ideal blankets of Table 12 by requiring each fusion neutron to produce 5% more tritium and correspondingly reducing F; the net fissile fuel production is then taken to be 85% of the resulting F. Table 14 pertains to blankets with neutron efficiency of 85%, both for tritium and fissile fuel production.

Table 12. Fissile fuel production ability of D-D-based and D-T systems having idealized blankets.

	Type of fusion neutron source			
		Cat-D	SCD	TCD
Parameter	D-T	D-T/D-D	D-T/D-D	D-T/D-D
ThO_2 volume fraction	8	25	25	16
T	1.0	0/0	0/0	0.4/0.2
F	1.08	1.88/.89	1.88/.89	1.53/.68
M	1.73	2.09/.70	2.09/.70	1.90/.60
Ratio D-D to D-T neutrons	0	1.17	1.17	0.57
Average Properties[a]				
$\overline{F}$	1.08	2.49	2.49	3.36
Fusion energy (MeV)	17.59	40.05	23.53	39.34
Total energy $\overline{W}$ (MeV)[b]	27.95	60.59	44.07	67.63
$\overline{F}/\overline{W}$	0.039	0.041	0.057	0.050
Normalized $\overline{F}/\overline{W}$	1.0	1.065	1.466	1.288

[a]Per $D(D,n)^3He$ reaction, in the D-D-based cases.
[b]Total energy deposited in the blanket plus charged fusion products energy.

Table 13. Fissile fuel production ability of D-D-based and D-T systems. Efficiency for tritium and fissile production is 95% and 85%, respectively.

Parameter	Type of fusion neutron source			
		Cat-D	SCD	TCD
	D-T	D-T/D-D	D-T/D-D	D-T/D-D
ThO_2 volume fraction	8	25	25	16
T net	1.0	0/0	0/0	0.4/0.2
T gross	1.05	0/0	0/0	0.42/.21
F gross	1.03	1.88/.89	1.88/.89	1.51/.67
F net	0.88	1.60/.76	1.60/.76	1.28/.57
M	1.73	2.09/.70	2.09/.70	1.90/.60
Ratio D-D to D-T neutrons	0	1.17	1.17	0.57
<u>Average properties</u>[a]				
$\bar{F}$	0.88	2.12	2.12	2.82
Fusion energy (MeV)	17.59	40.05	23.53	39.34
Total energy, $\bar{W}$ (MeV)[b]	27.95	60.59	44.07	67.63
$\bar{F}/\bar{W}$	0.031	0.035	0.048	0.042
Normalized $\bar{F}/\bar{W}$	1.0	1.116	1.534	1.330

[a]Per $D(D,n)^3He$ reaction, in the D-D-based cases.
[b]Total energy deposited in the blanket, plus charged fusion products energy.

Table 14. Fissile fuel production ability of D-D-based and D-T systems. Efficiency for tritium and fissile production is 85%. Tritium losses of 3%.

	Type of Fusion Neutron Source			
		Cat-D	SCD	TCD
Parameter	D-T	D-T/D-D	D-T/D-D	D-T/D-D
ThO_2 volume fraction	8	25	25	16
(F + T) ideal	2.08	1.88/.89	1.88/.89	1.93/.88
(F + T) net	1.77	1.60/.76	1.60/.76	1.64/.75
T gross	1.03	0/0	0/0	0.41/.21
F net	0.74	1.60/.76	1.60/.76	1.23/.54
Ratio D-D to D-T neutrons	0	1.17	1.17	0.57
Average Properties[a]				
$\bar{F}$	0.74	2.12	2.12	2.69
Fusion energy (MeV)	17.59	40.05	23.53	39.34
Total Energy, $\bar{W}$ (MeV)[b]	27.95	60.59	44.07	67.63
$\bar{F}/\bar{W}$	0.026	0.035	0.048	0.040
Normalized $\bar{F}/\bar{W}$	1.0	1.33	1.82	1.51

[a]Per $D(D,n)^3He$ reaction in the D-D-based cases.

[b]Total energy deposited in the blanket plus charged fusion products energy.

Such an efficiency is typical of many tokamak reactor designs (e.g., the STARFIRE design[12]). The blanket energy multiplication is assumed to be independent of the neutron, or blanket coverage efficiency to a first approximation.

Under all three sets of assumptions the SCD system offers the highest $\bar{F}/\bar{W}$, where the bars denote averages over the fusion neutrons spectrum, giving the highest support ratio. The second highest $\bar{F}/\bar{W}$ is offered by the TCD system, while the D-T system offers the lowest $\bar{F}/\bar{W}$. This ranking also turns out to be the descending order of difficulty in the confinement and ignition of the corresponding plasmas. Whether or not the benefit from the improved fissile fuel production ability could compensate for the more demanding, and hence, more expensive confinement and ignition requirements is, therefore, an important question for economic analysis.

The higher the blanket tritium production requirements are, the more sensitive the system is to the neutron efficiency for the production of tritium. Thus, the D-T system pays the highest penalty for the loss in the tritium production efficiency--its support ratio decreases by 33% going from the no-loss to 15%-loss scenario. Correspondingly, the relative merit of the D-D-based systems increases with the loss in the tritium production efficiency.

The $\bar{F}/\bar{W}$ values of Tables 12 to 14 are significantly lower than those of Table 1. The differences stem from the fact that the Table 1 values pertain to leakage-free, structure-free idealized blankets and assume equal probability for the $D(D,n)^3He$ and the $D(D,p)T$ reactions, no ^3He burn, and no neutron or tritium losses.

4.0. ECONOMIC ANALYSIS

4.1. INTRODUCTION

The results obtained in Sections 2 and 3 indicate that the D-D-based fusion breeders considered offer a wide range of fissile fuel production ability, fusion and blanket power densities, and power recirculation fraction. Moreover, we have seen that relative to D-T plasmas, D-D-based plasmas offer higher fissile fuel production per given thermal capacity, but require more demanding confinement conditions and offer a lower fusion energy gain and power density. Thus, an assessment of the prospects of the D-D-based FBs necessitates an economic analysis, properly weighting the merits against the drawbacks of the different FB concepts.

Three central questions involving the economic prospects of the fusion breeders are considered

(1) The relative promise of the various modes of operation of the tritium-assisted and D-D-based FBs.

(2) The prospects for the symbiotic system of tritium-assisted D-D-based FBs and LWRs to compete economically with LWRs supported by enrichment facilities.

(3) The promise of tritium-assisted, D-D-based FBs relative to the promise of D-T FBs.

A thorough assessment of the prospects of D-D-based vs D-T FBs requires a search for optimal designs for the different types of FBs. Such a search, which must include conceptual design of FBs, is beyond the scope of this work. Instead, we shall consider existing designs and economic analyses of fusion power reactors, convert these reactors into FBs, and make a rough estimate of their economics.

Six different D-D-based FBs are examined. Their fusion drivers use a Cat-D plasma; a tritium-assisted Cat-D plasma with $\gamma_{DT} = 0.4$ and $\gamma_{DD} = 0.2$; a SCD plasma, two TCD plasmas, one at 40 keV and one at 60 keV; and a tritium-assisted TCD plasma in which the fraction of the ^{3}He allowed to fuse (40%) is higher than the minimal. All of the blankets are the type studied in Section 3.

4.2. STRATEGY AND ASSUMPTIONS

The D-T FB is modeled after the STARFIRE tokamak,[12] whereas all the D-D-based FBs are modeled after the WILDCAT tokamak.[13] The STARFIRE-WILDCAT pair was chosen for the present analysis because their designs are relatively consistent, they represent the most recent generation of tokamak reactor designs, their economic analysis uses the standard DOE accounts for fusion reactor cost estimates,[22,23] and the economic studies are sufficiently detailed to enable us to adjust their cost when converting them into FBs.

The conversion of the STARFIRE and WILDCAT fusion reactors into a D-T and a Cat-D fusion breeder is straightforward: the original blankets are replaced by the beryllium fission-suppressed blanket of the appropriate composition (Section 3); the power levels are adjusted in accordance with the change in the blanket's energy multiplication; and the cost of the major components is scaled with the required power levels. In the case of the blankets that differ in composition, extra expenses are imposed on the FBs for special requirements associated with the handling of fissile fuel and fission products.

The conversion of WILDCAT to the PCD and tritium-assisted FBs is carried out using a similar procedure, with one addition: the properties of the fusion drivers, including the fusion power output, fusion energy gain, and neutron source density, are first adjusted for the alternate mode of operation involved. This adjustment is made using the results of Section 2, assuming that the confinement parameter $n\tau_E$ is fixed at its reference WILDCAT value--4.1×10^{21} m^{-3} s. With this procedure, it is possible to use the existing design of the WILDCAT plasma confinement system. The compensation for the different plasmas is done by adjusting the plasma heating rate (i.e., the power recirculation, which reflects the differences in the plasma's fusion energy gain).

In converting the reference blankets into the FB blankets, it is assumed that the FB blanket coverage efficiency is 85%. This is similar to the STARFIRE design value. No changes in the machine design or dimensions are made to accomodate the fuel producing blanket. Similarly, the fission-suppressed blanket design is not adjusted to the actual space available for blankets in the reference STARFIRE and WILDCAT designs. We believe that near-optimal fission-suppressed blankets could be incorporated in STARFIRE and WILDCAT without significantly penalizing their design as far as the reactor size and plasma confinement ability are concerned. However, detailed conceptual designs of the FBs are required to reliably assess the consequences of matching the fission-suppressed blankets with the STARFIRE and WILDCAT type drivers.

The relative economics of the different FBs depends on their size. The size of a FB can be measured in several different ways, including its total thermal power output or the actual size of the fusion driver and associated blanket-shield system (including plasma chamber, blanket, and confining magnets). The relative economics of the D-D-based FBs is compared here for a specific size driver (that of WILDCAT). A thorough investigation of the prospects of D-D-based FBs calls also for comparing the economics of D-D-based FBs designed to have the same thermal power level, which is likely to be a limiting factor on the size of the FBs. As the power densities offered by the D-D-based FBs vary over a wide range, such a comparison requires redesigning the fusion drivers for several of the modes of operation considered, an undertaking beyond the scope of this work.

Instead, we select a promising D-D-based driver based on the equal-size fusion driver comparison and compare its economics with that of the D-T FB, now on an equal thermal power basis. Toward this purpose, the power level of the D-T FB is slightly scaled down to avoid an economy-of-scale bias in its favor.

Two figures of merit are used for assessing the economic prospects of the FBs. One is the ratio of the net annual income from the sale of electricity and fissile fuel to the total annual expenses including actual operation and maintenance costs, as well as return on the capital investment. This figure of merit indicates the rate of return on the investment in the FB, when the cost of electricity (COE) and the cost of fissile fuel (COF) pertain to the FB-free energy economy. In a sense, it is a measure of the profitability of the FBs.

The other figure of merit is the levelized cost of fissile fuel (LCOF), defined as the price to be asked for the fissile fuel, that will enable the buyer to cover all the expenses associated with the operation of the FB when the COE is the market value. This assumes that electricity is providing part of the revenue. The LCOF and COE are not self-consistent in the sense that LWRs buying fissile fuel at the LCOF could generate their electricity at a cost lower than the nominal COE.

The cost of electricity is assumed (Refs. 5 and 12) to be 40 mills/kWh, whereas the price of the ^{233}U is taken to be (Ref. 5) \$75/g (see also Appendix B); the latter corresponds to a uranium ore cost of \$40/lb. Three different values for fuel processing and fabrication costs are considered: \$2/g, \$22/g, and \$42/g; they represent processing and fabrication costs of thorium-^{233}U fuel in molten salt, metal, and oxide forms (upper limit for the oxide), respectively.[5]

4.3. FUSION BREEDER CHARACTERISTICS

The characteristics of the FBs examined in the economic analysis are summarized in Tables 15 and 16. These characteristics are determined using the plasma properties of Table 17, which were obtained with the simple model described in Section 2, along with the fissile fuel production abilities of Table 18, showing values that were obtained as described in Section 3.

In estimating the annual electricity and fissile fuel production of Table 15, it is assumed that the plant capacity factor is 0.75 for the D-D-based FBs and 0.72 for the D-T FB. This difference reflects the more frequent replacements of the first-wall/blanket modules of the STARFIRE design. Other assumptions used for evaluating the FB characteristics of Tables 15 and 16 are described in the table footnotes.

Two sets of characteristics are given for the D-T FBs. Those in Table 15 pertain to the reference D-T FB, obtained by replacing the original STARFIRE blanket by our fission-suppressed blanket. The characteristics in Table 16 are deduced from those of Table 15, by scaling down the total thermal power output to 89% of the reference D-T FB to match the thermal power of the TCD1 FB, with which it is compared.

Observe in Table 15 that, of the D-D-based FBs, the tritium-assisted modes of operation offer both the highest electrical power output (the Cat-D-T case) and the highest fissile fuel production rate (the TCD1 case). All tritium-assisted D-D-based FBs offer a higher electricity and fissile fuel production rate than the Cat-D FB. For example, the TCD-α FB offers about 50% higher electricity and fissile fuel production rate than the Cat-D FB. Relative to the D-T FB, the TCD1 FB offers approximately 15% lower energy, but 47% higher fissile fuel production rate.

A somewhat distinct performance is offered by the SCD-FB. Its net efficiency for the production of electricity is very low--only about 4%. This is because of the relatively high recirculated power fraction, which in turn is caused by the low Q of the SCD plasma, and because of the relatively low blanket power. Even though the SCD mode of operation offers the highest support ratio (or F/W) of all fusion systems considered (see, for example, Table 18), its total fissile fuel production rate is lower than that of certain of the TCD FBs, such as TCD1 and TCD-α.

Table 15. Characteristics of the fusion breeders considered for the economic analysis.

Parameter	STARFIRE D-T	WILDCAT Cat-D	Cat-D-T	SCD	TCD1	TCD2	TCD-α
Fusion power (MW)	3510	2054	2782	1704	2710	1901	2743
Blanket power (MW)	2061	1054	1715	1483	1952	1323	1844
Total nuclear power (MW)	5571	3108	4497	3187	4662	3224	4587
Q	38.8	20	38.8[a]	3.5	9	15.5	19.5
Plasma heating and driving power (MW)	90	103	72	487	301	123	141
Electric power for heating[b] (MWe)	153	147	102	696	430	175	201
Nuclear + heating power (MW)	5724	3255	4600	2400	5092	3400	4788
Pumping power[c] (MWe)	47	27	38	20	42	28	39
Total thermal power (MW)	5771	3282	4638	2420	5134	3428	4827
Gross electricity generation[d] (MWe)	2060	1172	1656	864	1833	1224	1724
Recirculated power[e] (MWe)	272	228	204	764	539	259	306
Net electricity generation (MWe)	1788	944	1452	100	1294	965	1418
Net efficiency (%)	31.0	28.8	31.3	4.1	25.2	28.2	29.4
Annual production of electricity[f] (10^9kwh/yr)	11.28	6.205	9.544	0.66	8.505	6.344	9.324
Annual fissile fuel production[f] (kg/yr)	8102	6228	7677	8760	10615	7192	9403

[a] Assumed equal to that of the D-T plasma so as to allow for current drive.

[b] Assuming electric-to-plasma energy conversion efficiency of 70% after the STARFIRE and WILDCAT design values.[12,13]

[c] Assumed proportional to total thermal power to be removed, with proportionality constant of 0.0082 after the STARFIRE/WILDCAT design values.

[d] Assuming 35.7% thermal-to-electrical energy conversion efficiency after STARFIRE/WILDCAT design values.

[e] Consisting of the plasma heating power, pumping power (see both values in table), power for cryogenics (14 MWe), magnets (5 MWe), balance of plant (13 MWe), and power for heat transport and condensation systems (assumed proportional to total thermal gross electrical power; proportionality constant = 0.0107).

[f] Assuming a capacity factor of 0.72 and 0.75 for the D-T and D-D based FBs, respectively.

Table 16. Characteristics of a D-T fusion breeder adjusted to the power level of the TCD1 fusion breeder.

Parameter	Fusion breeder D-T STARFIRE	Fusion breeder TCD1 WILDCAT
Total thermal power (MW)	5,134[a]	5,134
Gross electricity generation (MWe)	1,833	1,833
Electric power for heating (MWe)	144	430
Pumping power (MWe)	42	42
Power for heat transport and condensation (MWe)	35	35
Total recirculated power[b] (MWe)	253	539
Net electricity generation (MWe)	1,580	1,294
Net efficiency (%)	30.8	25.2
Annual production of electricity[c] (10^9 kwh/yr)	9.97	8.505
Annual fissile fuel production[c] (kg/yr)	7,208	10,615

[a]Reduced to 89% of original value.

[b]Consisting of the electric power for heating, pumping power, power for the heat transport and condensation system (all these are listed in the table), power for cryogenics, magnets, and balance of plant.

[c]Assuming a capacity factor of 0.72 and 0.75 for the D-T- and D-D-based FBs, respectively.

Table 17. Characteristics of D-D-based and tritium-assisted plasmas for WILDCAT ($n_e\tau$ = 4.1 x $10^{21}m^{-3}s$).

Parameter	Plasma					
	Cat-D	Cat-D-T	SCD	TCD-1	TCD-2	TCD-α[c]
Number of tritons produced per:						
D-T neutron(γ_{DT})	0	0.4	0	0.4	0.4	0.4
D-D neutron (γ_{DD})	0	0.2	0	0.2	0.2	0.2
Fraction of ^{3}He fused	1.0	1.0	0.103	0.104	0.215	0.4
Plasma temperature (keV)	37.5	37.5	40	40	60	40
Fusion energy gain (Q)	20	∞	3.5	9	15.5	19.5
Relative ion density						
D	0.836	0.826	0.952	0.935	0.902	0.898
T	0.004	0.007	0.004	0.009	0.012	0.009
^{3}He	0.123	0.122	0.014	0.013	0.019	0.050
α	0.018	0.018	0.015	0.014	0.025	0.016
P	0.018	0.027	0.015	0.028	0.043	0.028
Relative fusion power density[a]	1.0	1.35	0.83	1.32	0.93	1.34
Charged/total fusion power	0.64	0.51	0.39	0.31	0.35	0.40
Neutron source density[a,b] (relative)						
D-D neutrons	1.0	0.97	1.41	1.34	0.93	1.20
D-T neutrons	1.0	1.99	1.41	2.74	1.80	2.45
D-T/D-D neutron ratio	0.856	1.762	1.170	1.756	1.655	1.750

[a]Total fusion power of the WILDCAT (Cat-D) fusion breeder is 2054 MW, and its neutron source intensity is 3.15 x 10^{20} neutrons per second.

[b]The first wall loading for the reference WILDCAT reactor is 0.5 MW/m^2 of D-T neutrons, 0.10 MW/m^2 of D-D neutrons, and 1.0 MW/m^2 of thermal energy.

[c]The ^{3}He left over from this plasma can provide only about 2/3 of the tritons needed. The other 1/3 will have to be produced from lithium or provided by the client fission reactors.

Table 18. Fissile fuel production ability of the FBs considered.

Parameter	D-T	Cat-D D-T/D-D	Cat-D-T D-T/D-D	SCD D-T/D-D	TCD1 D-T/D-D	TCD2 D-T/D-D	TCD-α D-T/D-D
			Type of fusion neutron source				
F net[a]	0.74	1.60/.76	1.23/.54	1.60/.76	1.23/.54	1.23/.54	1.23/.54
Fissile atoms per D-D-n		2.130	2.707	2.128	2.700	2.575	2.693
M	1.73	2.09/.70	2.09/.70	2.09/.70	1.90/.60	1.90/.60	1.96/.63
Energy added by blanket per D-D neutron (MeV)	10.33	20.61	34.57	20.59	28.36	27.07	30.19
Fusion energy per D-D neutron (MeV)	17.59	40.15	56.07	23.66	39.37	38.90	44.90
Total nuclear energy per D-D neutron (MeV)	27.92	60.76	90.64	44.25	67.73	65.97	75.09
$\overline{F}/\overline{W}$							
Absolute (at/MeV)	0.0265	0.0351	0.0299	0.0481	0.0399	0.0390	0.0359
Normalized		1.0	0.852	1.370	1.137	1.111	1.022
	1.0	1.325	1.128	1.815	1.506	1.472	1.355

[a]See Table 14.

4.4. CAPITAL COST ADJUSTMENT

Table 19 summarizes the adjustment in the capital cost of the various FBs considered here. Only cost components with non-negligible change are listed. The assumptions used for estimating the adjustment in these cost components are the following (the component numbers are the account numbers used in Refs. 12 and 13):

21. Structure and site facilities
    - 21.04 Cooling system structures: Proportional to the waste heat Pw.
    - 21.10 Fuel handling and storage building: For tritium-breeding D-D-based FBs--same as for STARFIRE.
    - 21.17 Ventilation stack: Added for all tritium-breeding D-D-based FBs--same as for STARFIRE.

22. Reactor plant equipment
    - 22.01.01 Blanket and first wall: The PCA stainless steel of the WILDCAT design is replaced by Be, ThO_2, Fe and, when necessary, Li_2O, the cost of which is assumed to be (after Ref. 5) 35, 250, 54, 35 and 80 $/kg, respectively. A similar adjustment is made for the STARFIRE reactor, replacing the $LiAlO_2$ with Be, ThO_2, and Li_2O at the optimal proportions (see Table 14). The original STARFIRE and WILDCAT designs do not always provide a blanket zone as thick as that used for the FB blankets under consideration (see Section 3). Nevertheless, it may be possible to design realistic blankets for the FBs that will perform similarly to the idealized blanket of Section 3--at least relatively, without greatly increasing the size and cost of the fusion driver and blanket.
    - 22.01.03 Magnets: Reduced somewhat arbitrarily, by $5 million for the adjusted D-T FB (Table 20) to account for the $\sim$10% reduction in its power level.

Table 19. Direct cost adjustment of the fusion breeders. (All costs are in 1980 millions of dollars).

Parameter	Fusion breeder						
	D-T[a]	Cat-D	Cat-D-T	SCD	TCD1	TCD2	TCD-α
Direct cost of reference reactor[b]	1726	2213	2213	2213	2213	2213	2213
Additional direct costs[c]							
21.04	2.2	0.8	3.8	-1.1	4.9	1.1	4.2
21.10			1.7		1.7	1.7	1.7
21.17		1.8	1.8	1.8	1.8	1.8	1.8
22.01.01	33.0	83.7	83.4	83.7	83.4	83.4	83.4
22.01.03	-5.0						
22.01.04				130.3	67.2	6.6	12.8
22.01.07				43.3	22.3	2.2	4.3
22.02.01	14.6	5.6	26.1	-7.5	33.6	7.8	28.9
22.05	30.0	30.0	36.9	30.0	36.9	36.9	36.9
22.06	0.6	0.2	1.0	-0.3	1.2	0.3	1.1
22.98	0.9	0.7	1.3	0.4	1.4	0.9	1.3
23.01 and 23.02	10.4	4.3	18.5	-6.3	23.2	6.0	20.3
23.03 and 23.04	18.4	6.7	31.3	-9.0	40.3	9.3	34.7
23.05 and 23.06	16.4	6.6	31.0	-8.9	39.9	9.2	34.4
24.0	6.4	1.8	8.0	-2.9	10.0	2.7	8.8
Total	127.9	142.2	244.8	253.5	367.8	169.9	274.6
Total direct cost	1854	2355	2458	2467	2581	2383	2488

[a]Total thermal power level adjusted to match that of the TCD1 fusion breeder (pertaining to the D-T data of Table 16).

[b]As per Refs. 12 and 13 for STARFIRE and WILDCAT.

[c]See description of cost components given in the text.

22.01.04 Radio frequency (rf) heating and current drive: Assumed porportional to rf power (Ref. 13 gives no scaling laws for this cost component). The system cost for the D-T and CAT-D-T FBs is taken to be that of the reference STARFIRE and WILDCAT designs, as both primarily use this system for current drive.

22.01.07 Power supply switching and electrical storage: The compressional Alfven-wave heating and the current drive component of this item are assumed to be proportional to the rf power.

22.02 Main heat transfer and transport

22.02.01 Primary cooling system: Proportional to total thermal power P_{th}.

22.05 Fuel handling and storage system: $6.9 million is added to all tritium-breeding D-D-based FBs, as in the STARFIRE design, to account for such items as tritium recovery, purification, and storage. In addition, $30 million is added to all FBs to account for the need to handle fissile fuel and fission products somewhat arbitrarily. This price does not include fuel processing and fabrication facilities.

22.06 Other reactor plant equipment

22.06.07 Closed loop coolant system: Proportional to $(P_{th})^{0.7}$. This scaling law was derived from the STARFIRE and WILDCAT data.

22.06.08 Standby cooling system: Same as 22.06.07.

22.98 Spare parts allowance: 2% of all cost components adjustments pertaining to items 22.01 through 22.07.

23. Turbine plant equipment

23.01 Turbine generator: Proportional to $(P_{th})^{0.5}$.

23.02 Main steam system: Same as above.

23.03 Heat rejection system: Proportional to P_w.

23.04 Condensing system: Proportional to P_w (α P_{th}).

23.05 Feedwater heating system: Proportional to P_{th}.

23.06 Other turbine plant equipment: Proportional to P_{th}.

24. Electrical plant equipment

24.05 Electrical structures and wiring containers: Assumed to scale like $P_{eg}^{1/3}$, where P_{eg} is the gross electric power generated. This scaling law gives the 90% difference between the WILDCAT and STARFIRE costs for this component.[13]

24.06 Power and control wiring: Same as above.

4.5. FUSION BREEDER ECONOMICS

Using the total direct capital cost arrived at in Table 19, the total capital cost of the FBs is evaluated in Table 20 in the "then-current" (i.e., 1986) dollars.[12,13] Following Refs. 12 and 13, we assume the following accounting:

Account 91. Construction facilities, equipment and services: 10% of direct capital cost.

Account 92. Engineering and construction management services: 8% of direct capital cost.

Account 93. Other costs: 5% of direct capital cost.

Account 94. Interest during construction: 31.63% of the direct capital cost plus above three additions (i.e., 1.23 times the direct capital cost). This assumes a construction period of 6 years.

Account 95. Escalation during construction: Adding 18.96% to 1.23 times direct capital cost, assuming, again, 6 years construction time.

Shown also in Table 20 is the "dollars-per-kilowatt-installed" for the different FBs. The D-T FB has a clear advantage over the D-D-based FBs, in terms of the $/kWe figure-of-merit. Next to the D-T FB come the Cat-D-T and TCD-α FBs, with about 50% higher $/kWe, whereas the worst is the SCD FB. The poor performance of the SCD FB is caused by its poor energy balance, i.e., a large energy recirculation, leaving very little electrical energy for sale (see Table 16). Among the WILDCAT based FBs, the $/kWe figure-of-merit largely reflects the energy production ability of the different breeders, in terms of the total energy generated and its conversion efficiency to electrical energy for sale.

Table 20. Capital cost of fusion breeders in 1986 millions of dollars.

Parameter	Fusion breeder						
	D-T[a]	Cat-D	Cat-D-T	SCD	TCD1	TCD2	TCD-α
Total direct cost[b]	1854	2355	2458	2467	2581	2383	2488
91	185	236	246	247	258	238	249
92	148	188	197	197	206	191	199
93	93	118	123	123	129	119	124
Subtotal	2280	2897	3023	3034	3175	2931	3060
94	721	916	956	960	1004	927	968
95	432	549	573	575	602	556	580
Total capital cost	3433	4363	4552	4569	4781	4414	4608
$/kWe[c]	2135	4622	3135	45690	3695	4574	3250

[a] Total thermal power level adjusted to match that of the TCD1 fusion breeder according to the D-T data of Table 16.

[b] See description of capital cost adjustments in the text.

[c] Pertaining to net electric power.

The FB annual cost of operation is evaluated in Table 21, following the methodology and assumptions of Refs. 12 and 13:

1. The levelized capital cost is taken to be 15% of the total capital cost.
2. Operation and maintenance costs: The reference value of \$19.4 million/yr (the same for STARFIRE and WILDCAT) is increased somewhat arbitrarily by 20% to account for extra work that might be imposed by the presence of fissile fuel and fission products in the FBs. The result is multiplied by $(1 + 0.05)^6 = 1.34$, to account for the escalation rate.
3. Scheduled component replacement costs: This cost is adjusted in accordance with the cost of the blanket and rf system, and with the first-wall/blanket lifetime. This lifetime, evaluated in Table 22, is a function of the first-wall loading. The reference STARFIRE design is subjected to a thermal and neutron wall loading of $0.9 MW/m^2$ and $3.6 MW/m^2$, respectively. The resulting first-wall lifetime is estimated to be 6 years, at a load factor of 0.75. The corresponding reference WILDCAT values are $1 MW/m^2$, $0.5 MW/m^2$, and $0.1 MW/m^2$ for thermal wall loading, 14-MeV neutron wall loading, and 2.45-MeV neutron wall loading, respectively. The first-wall lifetime is estimated to be 20 years dictated by the lifetime of the first-wall beryllium coating; in the area of neutron radiation damage, the WILDCAT first-wall lifetime could be almost doubled.

   Table 22 compares the first-wall loading of the fusion breeders under consideration. Shown also in this table is the time it takes the first-wall to reach an integrated neutron loading of 16.2 $MW \cdot yr/m^2$, corresponding to the 6 years lifetime of the reference STARFIRE design. It is observed that only the TCD1 and TCD-α FBs first-wall loadings are less than 20 years.

   Two types of components that contribute to the scheduled component replacement cost are considered: the first wall/blanket system and the components associated with the heating and current drive system. The blanket cost additions that result from

Table 21. Annual cost of fusion breeders in 1986 millions of dollars per year.

Parameter	Fusion breeder						
	D-T[a]	Cat-D	Cat-D-T	SCD	TCD1	TCD2	TCD-α
Levelized capital cost	515.0	654.5	682.8	685.4	717.2	662.1	691.2
Operation & maintenance	31.2	31.2	31.2	31.2	31.2	31.2	31.2
Scheduled component replacement	27.8	15.9	15.6	18.9	17.4	16.0	16.2
Fusion fuel (D) costs	0.40	0.54	0.61	0.64	0.72	0.50	0.68
Total annual cost	574.4	702.1	730.2	736.1	766.5	709.8	739.3

[a] Total thermal power adjusted to 5134 MW.

Table 22. First wall loading, lifetime and component replacement cost of fusion breeders.

Parameter	Fusion Breeder						
	D-T[a]	Cat-D	Cat-D-T	SCD	TCD1	TCD2	TCD-α
First wall loading (MW/m^2)							
Thermal	0.82	1.0	1.08	0.51	0.64	0.51	0.84
14.1-MeV neutrons	3.2	0.5	0.995	0.705	1.37	0.90	1.225
2.45-MeV neutrons	--	0.1	0.097	0.014	0.13	0.09	0.120
Total neutrons	3.2	0.6	1.09	0.72	1.50	1.00	1.35
Time for integrated neutron fluence of 16.2 MWY/m^2	7.0	36.0	19.8	30.0	14.4	21.6	16.0
Heating and current drive components replacement costs ($M/yr)		0.60	0.42	2.84	1.76	0.71	0.82

[a] Total thermal power adjusted to 5134 MW. Load factor is 0.72 (vs 0.75 of the rest of the FBs).

converting the fusion power reactors into FBs are given in Table 19. The heating and current drive-scheduled-components replacement-cost estimates are given in Table 22; they are deduced from Ref. 13, scaling the reference Cat-D values proportional to the rf-system power-level. A 1.34 multiplier is applied to the overall component costs to account for cost esclation.

4. Fusion fuel cost: Accounting for the cost of the deutrium consumed, adjusted to account for the reduced fusion power of the D-T FB, and for the differences in the rate of the deuterium consumption for the WILDCAT based FBs.

Table 23 compares the overall economics of the FBs. The revenues from the sale of electricity pertain to 40 mills/kWh (see Section 4.2 and Appendix B). The revenues from the sale of fissile fuel are evaluated using one of the following approaches pertaining to the two figures-of-merit considered: for calculating the net annual income/total annual cost figure-of-merit, the value of the fissile fuel is taken to be \$75/g minus the processing and fabrication (P&F) costs. The P&F costs are assumed, after Ref. 5, to range in between \$2 to \$42 per gram of ^{233}U, depending on the fuel form and reprocessing capacity (see Section 4.2). The levelized cost of fissle fuel figure of merit is defined to be

$$\frac{\text{Total annual costs-revenues from sale of electricity}}{\text{Annual fissile fuel production}} + \text{P\&F costs.}$$

Considering the profitability figure of merit (i.e., net income to total cost), we observed that if the fuel could be processed and fabricated at the lowest cost conceivable, all but the SCD FBs considered can produce fissile fuel profitably. This implies that the corresponding FB-LWR symbiotic system might compete economically with the conventional LWR-enrichment plant system, even with the nominal value of uranium ore (\$40/lb U_3O_8).

Most promising of the D-D-based reactors are the TCD1 and TCD-α FBs; their profitability is about 25% lower than that of the D-T FB, but remarkably high in absolute terms. Comparing the tritium-assisted with tritium-assisted-free D-D-based FBs, it is clear that the tritium-assisted modes of operation offer a distinct economic advantage over the tritium-assisted-free D-D-based modes. Moreover, the partially catalyzed deuterium plasmas offer a clear economic advantage over the fully catalyzed mode of operation.

Table 23. Overall economics of the fusion breeders in 1986 millions of dollars per year.

Parameter	Fusion breeder						
	D-T[a]	Cat-D	Cat-D-T	SCD	TCD1	TCD2	TCD-α
Total annual cost	574.4	702.1	730.2	736.1	766.5	711.4	739.2
Gross annual income							
from electricity	398.8	248.2	381.8	26.3	340.2	253.8	373.0
from fuel	237.9[b] / 526.2	205.5 / 454.6	253.3 / 560.4	289.1 / 639.5	350.3 / 774.9	237.3 / 525.0	310.3 / 686.4
Net annual income	62.3 / 350.6	-248.4 / 0.70	-95.0 / 212.1	-420.7 / -70.3	-76.0 / 348.6	-220.3 / 67.4	-55.9 / 320.2
Net annual income / Total annual cost	0.109 / 0.610	-0.354 / 0.001	-0.130 / 0.291	-0.572 / -0.096	-0.099 / 0.455	-0.310 / 0.095	-0.076 / 0.433
Levelized cost of fissile fuel ($/g)	66.4 / 26.4	114.9 / 74.9	87.4 / 47.4	12.0 / 83.0	82.2 / 42.2	105.6 / 65.6	80.9 / 40.9

[a] Total thermal power adjusted to 5134 MW, as of TCD1 FB.

[b] In the notation (x/y), x stands for the value of the parameter for fuel processing and fabrication costs of $42/g, and y denotes those fuel costs at $2/g.

In the extreme of high P&F costs, none of the D-D-based FBs appears profitable, although the TCD1 and TCD-α FBs are very close to breakeven.

A similar picture is exhibited by the levelized fissile fuel figure-of-merit. Of the D-D-based FBs, the lowest cost of fissile fuel is being offered by the TCD-α and TCD1 FBs; except for P&F costs near their upper limit, this levelized cost of fissile fuel is lower than the value of fissile fuel in the present FB-free nuclear energy economy. Had the total thermal power of the D-T FB been adjusted to that of the TCD-α FB, the difference between the levelized cost of fissile fuel offered by these two FBs would have been even smaller than that of Table 23.

4.6. SENSITIVITY ANALYSIS

To get an indication of tne sensitivity of the D-D-based FB economics to key assumptions made in the analysis, the performance of a number of these FBs is re-evaluated using three alternate sets of assumptions:

1. Lower capital cost for the D-D-based FBs,
2. Higher fusion energy gain (Q) for the D-D-based FBs,
3. Higher fissile fuel value.

The economic consequences of two additional alterations, the sale of ^3He and tritium assistance from client fission reactors, are considered in Section 5.2 and 5.4, respectively.

4.6.1. Capital Cost Reduction

The capital cost of the reference WILDCAT design is about 30% higher than that of the reference STARFIRE.[12,13] In principle, it seems feasible to design more compact and cheaper D-D-based fusion drivers than WILDCAT, thus improving the economics of the corresponding FBs. For example, the capital cost of the recently designed Cat-D Compact Reversed Field Pinch Reactor (CRFPR) is only about 20% higher than that of the D-T CRFPR.[24] Also, the tritium-assisted D-D-based compact tokamaks, recently designed by the M.I.T. group,[25,26] offer another approach that might result in a lower capital cost compared with the capital cost of the corresponding D-T compact tokamak; the size and cost of the latter is likely to be first-wall-loading limited.

For the sake of the sensitivity study, it is assumed that the D-D-based FBs use an improved WILDCAT, the capital cost of which is higher than that of STARFIRE by only one-half the difference between the reference WILDCAT and STARFIRE. That is, the direct capital cost of the improved WILDCAT is assumed to be $1970 million in 1980 dollars, versus $1726 million and $2213 million for the reference STARFIRE and WILDCAT (See Table 19).

Table 24 compares the economics of the resulting reduced cost FBs with that of the D-T FB. We observe that the profitability of the TCD1 FB can almost match that of the D-T FB, and that the levelized cost of the TCD-α produced fissile fuel can be within 10% of that produced in the D-T FB; indeed, it falls below the value of fissile fuel in the LWR-enrichment system.

4.6.2. Fusion Energy Gain Improvement

If the fusion energy gain of the TCD1 FB proves to be 20 (rather than 9 as assumed so far), how much will the economic prospects for this FB improve? This postulated improvement in the fusion energy gain (Q) may come about as a result of better than assumed energy confinement time, reduced cyclotron radiation losses, and perhaps by the use of polarized plasma.[27]

Table 25 compares the economics of the high Q with the nominal Q TCD1 FBs. The total thermal power of the high Q TCD1 FB is increased by approximately 5%, to match that of the reference TCD1 FB. Thus, only a minor adjustment in the capital and annual costs of the FB is required (mostly consisting of scheduled component replacements). We observe that the TCD1 FB economics are not very sensitive to the FB fusion energy gain increase.

4.6.3. Increased Price of Uranium

If the uranium ore price increases from the $40/lb assumed for the present analysis to $80/lb of U_3O_8, how will this effect the economic prospects of the D-D-based FBs? Using the data of Appendix B with $80/lb U_3O_8, the COE from the conventional LWR-enrichment facilities is estimated to increase by about 3.5 mills/kWh--to 43.5 mills/kWh. The value of fissile fuel is estimated to increase to approximately $115/g.

Table 26 compares the profitability of the FBs with the altered costs of electricity and fissile fuel. We observe that the profitability of all the FBs is much higher by at least a factor of two than that of the reference FBs

Table 24. Economics of D-D-based fusion breeders based on reduced cost WILDCAT, in millions of 1986 dollars per year.

Parameter	Fusion breeder						
	D-T[a]	Cat-D	Cat-D-T	SCD	TCD1	TCD2	TCD-α
Total annual cost	574.4	634.6	662.6	568.6	699.0	643.9	671.7
Net annual income[b]	62.3[c] / 350.6	-180.9 / 68.2	-27.5 / 279.6	-353.2 / -2.8	-8.5 / 416.1	-152.8 / 134.9	11.6 / 387.7
Net annual income / Total annual cost	0.109 / 0.610	-0.285 / 0.108	-0.042 / 0.422	-0.528 / -0.004	-0.012 / 0.595	-0.237 / 0.210	0.017 / 0.577
Levelized cost of fissile fuel ($/g)	66.4 / 26.4	104.0 / 64.0	78.6 / 38.6	115.3 / 75.3	75.8 / 35.8	96.2 / 56.2	73.8 / 33.8

[a]Same as in Table 23.

[b]The gross and net annual incomes are the same as in Table 23.

[c]See footnote b, Table 23.

Table 25. Effect of fusion energy gain on the TCD1 FB economics, in 1986 millions of dollars.

Parameter	TCD1 Fusion breeder: Nominal[a] Q	TCD1 Fusion breeder: High Q
Fusion energy gain, Q	9	20
Fusion power (MW)	2710	2842
Total nuclear power (MW)	4662	4889
Electric power for heating (MWe)	430.2	194
Pumping power (MWe)	41.8	40
Total power for electricity (MW)	5134.1	5134
Gross electricity generation (MWe)	1833	1833
Recirculated power (MWe)	539	301
Net electricity generation (MWe)	1294	1532
Net efficiency (%)	25.2	29.8
Annual production		
of electricity (10^9 kWh/yr)	8505	8919
of fissile fuel (kg/yr)	10,615	11,132
Levelized capital cost	717.2	706
Scheduled component replacement cost	17.4	16
Total annual cost	766.5	754
Gross annual income		
from electricity	340.2	356.8
	350.3[b]	367.4
from fuel	774.9	812.6
$\frac{\text{Net annual income}}{\text{Total annual cost}}$	$\frac{-0.099}{0.455}$	$\frac{-0.039}{0.551}$
Levelized cost of fissile fuel ($/g)	$\frac{82.2}{42.2}$	$\frac{77.7}{37.7}$

[a] Values taken from Tables 15 and 21.

[b] See footnote b, Table 23.

Table 26. Economics of the fusion breeders with increased uranium price,[a] in millions of 1986 dollars per year.

Parameter	Fusion breeder D-T[a]	Cat-D	Cat-D-T	SCD	TCD1	TCD2	TCD-α
Total annual cost	574.4	702.1	730.2	736.1	766.5	711.4	739.2
Gross annual income from electricity	433.7	269.9	415.2	28.6	370.0	276.0	405.6
	526.2	454.6	560.4	639.5	774.9	525.0	686.4
from fuel	814.5	703.8	867.5	989.9	1199.5	812.7	1062.5
Net annual income	385.5	22.4	245.5	-68.0	378.4	89.6	352.8
	673.8	271.6	552.6	282.4	803.0	377.3	728.9
Net annual income	0.671	0.032	0.336	-0.092	0.494	0.126	0.477
Total annual cost	1.173	0.387	0.757	0.384	1.048	0.530	0.986

[a] U_3O_8 price assumed to be \$80/lb. See additional comments in Table 23.

in Table 23. In fact, all the FBs considered are estimated to be profitable; an insignificant exception is the SCD FB when the P&F costs are at their extreme high value. The economic prospects of the TCD1 FB is very similar to that of the D-T FB.

4.6.4. Discussion

Based on the limited alternative scenarios considered, we can conclude that the economics of the tritium-assisted PCD FBs can closely approach that of the D-T FB. If it turns out that two or more of improvements/alterations can be realized simultaneously, the economics of the more promising D-D-based FBs, notably of the TCD1 and TCD-α FBs, may even surpass that of the D-T FB. In any case, tritium-assisted D-D-based FBs clearly show an economic viability.

Moreover, the tritium-assisted PCD FBs have a number of unique merits that were not factored into the economic analysis. These merits include

1. High support ratio,
2. A source of ^{3}He,
3. Simpler and safer blanket design.

5.0. OTHER CONSIDERATIONS

5.1. SUPPORT RATIO

The fissile fuel production ability and, consequently, the support ratio of the D-D-based FBs can be significantly higher than that of D-T FB. For example, the support ratios of the most economical D-D-based FBs considered are approximately 40% (TCD-α) and 50% (TCD1) higher, respectively, than that of the D-T FB, as deduced from the data of Table 15. Two implications of this phenomenon are noteworthy:

1. The D-D-based FBs are more attractive fuel factories for safeguarded fuel centers because they enable us to reduce the number of sites needed to provide the fuel needs of the fission reactors, or the total thermal capacity that has to be installed in and disposed from

a given site. Thus, as long as the economics of D-D-based and D-T FBs is not too different, the higher support ratio is a definite advantage favoring the D-D-based FBs.

2. Because of economy-of-scale of fuel-treating facilities, the reprocessing and refabrication costs of the D-D-based FB fuel may be lower than those of the D-T FB. However, no allowance for this difference was made in the analysis.

5.2. A SOURCE OF ^{3}He

As far as the neutron balance is considered, lithium could be substituted for the ^{3}He used in the TCD mode of operation to provide the tritium necessary for attaining the same ratio of D-T to D-D neutron source intensities and the same plasma properties as in the reference TCD plasma. The lithium could be located either in a narrow zone adjacent to the first wall, replacing the ^{3}He in the TCD reactor, or distributed in the blanket. Because it is not necessary to breed tritium, there is much more flexibility in the design of the lithium system in a tritium-assisted SCD, (i.e., SCD-T) reactor than in a D-T reactor.

The use of ^{6}Li instead of ^{3}He is somewhat less favorable, in terms of the energy balance, because the binding energy released per neutron capture in ^{6}Li is 4.78 MeV vs 0.76 MeV in ^{3}He. This difference is responsible, however, for only a few percent reduction in the support ratio.

The ^{3}He spared in the SCD-T or SCD modes of operation could provide the basis for a D-^{3}He (i.e., clean) fusion power economy.[9] The sale of ^{3}He could possibly increase the revenues of the SCD-T FB, thus improving its economics. Consider, for example, the TCD1 FB, using lithium instead of ^{3}He for tritium production in its blanket. With the resulting reactor, it is possible to spare about 45 kg of ^{3}He per year.

Without knowing the economics of the D-^{3}He power reactors, it is difficult to assign a value to the FB-produced ^{3}He. As an indication of the contribution of the revenues from the sale of ^{3}He to the economics of the SCD-T FB, we assume that the D-^{3}He fusion power reactors can pay as much as 20% of their COE for the purchase of ^{3}He--the same fraction paid by the LWR for its fuel. Realizing that one ^{3}He atom generates 18.35 MeV of thermal energy, which is converted to electricity at 50% efficiency (possibly using direct energy conversion), and assuming that the COE is 40 mills/kWh, the value of ^3He is found to be $452/g. This is below the cost that ^3He could be purchased for today--$735/g,[28] and considerably below the cost of production of ^3He in present day production reactors, assumed to be about $7500/g, which is the production cost of tritium.[28]

Table 27 shows the effect of possible revenues from the sale of ^3He on the economics of the SCD-T fusion breeder. We observe that with the price of ^3He close to its high limit, the economics of the SCD-T FB surpasses that of the D-T FB (see Table 23). Near its low price range, ^3He can improve the SCD-T fusion breeder economics by only a few percent.

5.3. LITHIUM-FREE BLANKETS

The D-D-based FBs can be designed to be free of lithium, thus simplifying the blanket design and improving the reactor safety.

It appears that incorporating the ^3He-to-tritium conversion system in the blanket of TCD reactors could be done without many difficult engineering and safety problems. Because of its very high cross section for the capture of low energy neutrons (more than 5,000 barns for 2200 m/s neutrons) only a relatively small amount of ^3He is required to make an efficient ^3He-to-tritium converter. For example, if distributed across the blanket, the amount of ^3He needed to support the TCD modes of operation considered is approximately 5 moles per-square-meter of first wall area. This is almost 40 times lower than the number of moles of Li_2O (using natural lithium) needed to provide the same tritium production rate. This ^3He could also serve as the blanket coolant, thus further simplifying the blanket design.

Alternatively, if concentrated near or within the design of the first wall, the amount of ^3He needed is approximately 35 moles per-square-meter of first wall area--about seven times more than with the uniform distribution across the blanket. This is still a much smaller amount than lithium required to provide the same tritium breeding. In this arrangement, the ^3He could be part of the first wall cooling system.

Because helium is inert, there are neither safety hazards, nor severe material compatibility problems associated with its use. Moreover, being gaseous and of a relatively small quantity, the ^3He system could be

Table 27. Effect of revenues from ^{3}He on the economics of the SCD-T fusion breeder, in 1986 millions of dollars.[a]

Parameter	Price of ^{3}He (\$/g): 0	500	5000
Total annual cost	766.5	766.5	766.5
Gross annual income			
from electricity	340.2	340.2	340.2
from fissile fuel	$\frac{350.3}{774.9}$	$\frac{350.3}{774.9}$	$\frac{350.3}{774.9}$
from ^{3}He	0	22.3	223
Net annual income	$\frac{-76.0}{348.6}$	$\frac{-53.7}{370.9}$	$\frac{147.0}{571.6}$
$\frac{\text{Net annual income}}{\text{Total annual cost}}$	$\frac{-0.099}{0.455}$	$\frac{-0.071}{0.484}$	$\frac{0.192}{0.746}$
Levelized cost of fissile fuel (\$/g)	$\frac{82.2}{42.2}$	$\frac{80.1}{40.1}$	$\frac{61.2}{21.2}$

[a]See comments in Table 23.

maintained with the aid of on-line processing at a relatively low tritium concentration and overall inventory.

The amount of 5 moles/m^2 of ^3He inventory needed for starting the TCD reactors could be accumulated in approximately 4 months of operation of the TCD drivers. Extra ^3He will be required for out-of-blanket inventories. Alternatively, it is not unlikely that the initial inventory of ^3He needed for the TCD reactors could be obtained at an acceptable cost from the waste (i.e., radioactive decay product) of the tritium stockpiles accumulated by the time the FBs will be ready for start up.

5.4. TRITIUM ASSISTANCE BY CLIENT REACTORS

Rather than producing all the tritium needed for the tritium-assisted modes of operation considered in the blankets of the FBs, it is possible to generate part or all of it in the client fission reactors (LWRs, in the scenarios considered). A variety of avenues exist for doing so--for example, those cited in Refs. 29 to 31.

Especially relevant to the tritium assistance mode of operation under consideration is the EPRI sponsored study[31] that examines the prospects of designing D-T hybrid reactors to be free of tritium production; the tritium for these hybrids is to be provided by fission reactors. A primary conclusion of this study is that, even though the hybrid reactor blankets could be simplified by designing them to be used without tritium, the overall neutron economy and the economics of the hybrid-based nuclear system are not likely to improve; the penalty paid by the fission reactors in increased enrichment, reduced breeding ratio, etc. may be too high to pay.

We propose yet another approach for using fission reactors to provide tritium to the FBs. This concept[11] utilizes neutrons otherwise lost by parasitic captures in the fission reactors. This can be done, for example, by modifying the control elements, including the control rods, regulating rods, burnable poisons and possibly soluble poisons of these reactors to contain ^6Li or ^3He as the primary neutron absorber. Such modifications can probably be done without impairing the performance of the LWRs, and with a relatively small expenditure. Consequently, for this very preliminary analysis we will assume that the tritium by-product from the LWRs is provided to the FBs free of charge.

Another difference between the tritium assistance mode of operation proposed here and the no-tritium approach of Ref. 31 is that the latter calls for providing, from the fission reactors, one triton (net) per fusion neutron. In contrast, the tritium-assisted modes of operation under consideration here need only a fraction of a triton (0.2 to 0.6) per fusion neutron, and they involve D-D-based fusion.

5.4.1. Degree of Tritium Assistance

The average number of tritons that can be produced in the client fission reactors per fusion neutron of type x (x = DD or DT) is denoted here by γ_x. This quantity can be estimated from the expression,

$$\gamma_x = F_x \frac{\delta \nu}{(1 - CR)\,(1 + \alpha)} \quad , \tag{1}$$

which is based on a simplified neutron balance model. In the above, F_x is the net number of fissile atoms produced in the FB per fusion neutron of type x; ν and δ are the average number of neutrons born per fission reaction and the fraction of these neutrons that can be devoted to the production of tritium in the fission reactor, respectively; CR is the average conversion ratio of these reactors, and α is the average capture-to-fission ratio of the fissile fuel used.

Based on neutron balance or reactivity data of contemporary LWRs,[32] it is estimated (see Appendix C) that on the average as many as 5% of the fission neutrons born in LWRs of both the PWR and the BWR type might be utilized for tritium production without impairing the performance of these reactors. For LWRs operating on the Th/^{233}U fuel cycle, CR $\sim$ 0.7, α = 0.1 and ν = 2.50. With a δ of 5% this gives

$$\gamma_x = 0.38\, F_x \quad . \tag{2}$$

If advanced converter reactors are used instead of the conventional LWRs, it will be possible to significantly increase the degree of tritium assistance available from the client reactors. For example, consider the high conversion ratio LWRs operating on the U/Pu fuel cycle, such as those of Refs. 33 and 34. The conversion ratio of these advanced PWR (APWR) designs is estimated[33,34] to be approximately 0.9. Assuming that the fraction of fission neutrons available for tritium production in the APWR is half that of conventional LWRs (i.e., $\delta \sim 2.5\%$) and that for ^{239}Pu, $\nu = 2.9$ and $\alpha = 0.37$, we obtain, from Eq. (1)

$$\gamma_x = 0.53\ F_x\ , \tag{3}$$

implying that half of the tritium needed for running the FBs might be supplied by the client fission reactors.

5.4.2. Effect on Fusion Breeder Performance

The tritium produced in the client reactors can assist the performance of the FBs in several ways, including: (1) reducing the number of tritons that need to be produced per fusion neutron in the blanket of the FB or even eliminating it all together, thus enabling us to increase the fissile fuel production ability of these FBs; (2) increasing the number of D-T to D-D reactions in the plasma of D-D-based FBs, thus improving the plasma fusion energy gain, power, as well as neutron source density.

For illustration consider the TCD1 FB that produces 0.4/0.2 tritons per DT/DD neutron (converting its ^{3}He into tritium) and is also provided with 0.2/0.1 tritons from its client reactors; this degree of tritium assistance corresponds to γ_x values that are approximately half the value given by the expression of Eq. (2). Selected characteristics of the resulting tritium-assisted TCD, or TCD-T, mode of operation are compared in Table 28 with those of the reference TCD1 FB, as well with the corresponding tritium-assisted D-T FB. The latter uses the client reactor produced tritium to upgrade its fissile fuel production ability.

The assumed tritium assistance from the client reactors increases the TCD FB fusion energy gain from 9 to 31, and the fusion power density by about 43%; its fissile fuel production ability as measured in terms of F/W, or the support-ratio, is essentially unchanged. In case of the D-T FB, the client reactor's tritium assistance improves the fissile fuel production ability by about 20%, without effecting the FB energy balance or power density.

Table 28. Effect of tritium assistance from client reactors on the performance of fusion breeders.

Parameter	Fusion breeder: TCD1 (reference)	TCD-T	D-T
Number T from blanket per neutron	0.4/0.2[a]	0.4/0.2	0.75
Number T to plasma per neutron	0.4/0.2	0.6/0.3	1.0
F_o	1.23/.54	1.23/.54	0.74
F	1.23/.54	1.23/.54	0.89
$\bar{F}$ (per D-D neutron)	2.70	4.08	
Fusion energy[b] (MeV)	39.37	42.66	17.59
Total nuclear energy,[b] $\bar{W}$ (MeV)	67.73	101.88	27.92
$\bar{F}/\bar{W}$	0.0399	0.0401	0.0319
Fusion power density[c] (relative)	1.32	1.88	
Charged/total fusion power	0.31	0.27	
Neutron source density (relative)			
D-D neutrons	1.34	1.27	
D-T neutrons	2.74	4.29	
D-T/D-D neutron ratio	1.76	2.88	
First wall loading (MW/m^2)			
Thermal	0.64	0.57(0.47)[d]	0.82
14.1-MeV neutrons	1.37	2.15(1.80)	3.2
2.45-MeV neutrons	0.13	0.12(0.10)	-
Total neutrons	1.50	2.27(1.90)	3.2

[a] In the notation x/y, x is for to D-T neutrons and y for D-D neutrons.

[b] Per $D(D,n)^3He$ reaction, for the TCD FBs, and per D-T reaction in the case of the D-T FB.

[c] See comments in Table 17.

[d] In the notation x(y), x stands for the nominal value and y is the value adjusted with the power level (i.e., reduced by 19.3%).

Table 28. (Continued).

Parameter	Fusion Breeder		
	TCD1 (reference)	TCD-T	D-T
Fusion energy gain, Q	9	31.3	38.8
Fusion power (MW)	2710	3866(3241)[d]	3510
Total nuclear power (MW)	4662	6651(5576)	5771
Electric power for heating (MWe)	430	176(148)	153
Pumping power (MWe)	42	56(47)	47
Total thermal power (MW)	5134	6883(5771)	5771
Gross electric power (MWe)	1833	2457(2060)	2060
Total recirculated power (MWe)	539	312(267)	272
Net electric power (MWe)	1294	2145(1793)	1788
Annual production of			
electricity (10^9 kWh/yr)	8,505	14,100(11,787)	11,280
fissile fuel (kg/yr)	10,615	15,228(12,766)	9,744
Direct cost of reference reactor[e]	2213	2213	1726
Additional direct cost	367.8	319.0[f]	191.1
Total capital cost (1986 $M)	4781	4690	3551
Annual costs (1986 $M/yr)			
Levelized capital cost	717.2	703.5	523.7
Operation and maintenance	31.2	31.2	31.2
Scheduled component replacement	17.4	27.2	30.6
Fusion fuel cost (deuterium)	0.72	0.83	0.45
Total	766.5	762.7	586.0

[d] In the notation (x/y), x stands for the nominal value and y is the value adjusted with the power level (i.e., reduced by 19.3%).

[e] The direct costs are given in 1980 millions of dollars.

[f] All TCD-T cost values are for to the adjusted power level.

Table 28. (Continued).

Parameter	Fusion breeder		
	TCD1 (reference)	TCD-T	D-T
Gross annual income			
from electricity	340.2	451.2	451.2
from fuel	350.3[g]/774.9	421.3/931.9	321.6/711.3
Net annual income	-76.0/348.6	109.8/620.4	186.8/576.5
Net annual income / Total annual cost	-0.099/0.455	0.144/0.813	0.319/0.984
Levelized cost of fissile fuel ($/g)	82.2/42.2	66.4/26.4	55.8/15.8

[g] In the notation x/y, x denotes the upper limit of the FBs fuel processing and fabrication costs ($42/g ^{233}U) and y to the lower limit ($2/g).

In calculating the effect of tritium assistance from client reactors on the fuel production ability of the FBs, it is necessary to account for the coupling between the γ_x and F_x values as shown by Eq. (2). Thus, when γ_x substitutes for part of the FB blanket produced tritium, the FB neutrons spared can be used to augment F_x, giving

$$F_x \simeq F_{xo} + \gamma_x \quad , \tag{4}$$

where F_{xo} is the value of F_x without tritium assistance by the client reactors or by the external source of tritium in general. Substituting Eq. (4) into Eq. (1) gives

$$\gamma_x = \frac{C}{1 - C} F_{xo} \tag{5}$$

$$F_x = \frac{1}{1 - C} F_{xo} \quad , \tag{6}$$

where $C \equiv \delta\nu/[(1 - CR)(1 + \alpha)]$. For the case of the D-T FB under consideration, $F_o = 0.74$, $C = 0.17$, and, therefore, $F \simeq 0.89$.

The power level of the TCD-T FB is found to be significantly higher than that of the D-T FB. To avoid an economy-of-scale bias in favor of the TCD-T FB, the power level of the latter was lowered by 19.3%. The capital, as well as scheduled component replacement and fusion fuel costs of the TCD-T and D-T FBs were adjusted in accordance with their power level and first wall/blanket lifetime, following the assumptions and procedures described in Section 4.

Comparing the economics of the client reactors tritium-assisted FBs with that of the reference FBs (Table 23), we see that even the relatively low degree of tritium-assistance considered here significantly improves the profitability of the FBs (net annual income/total annual cost), and noticeably reduces the levelized cost of fissile fuel. The economic effect of this tritium assistance is somewhat more pronounced for the TCD than for the D-T FB, thus narrowing the economic margin in favor of the D-T FB.

6.0. CONCLUSIONS AND RECOMMENDATIONS

Limited as it is in the scope of systems and design details considered, the results of this investigation do not allow firm quantitative conclusions on the properties of D-D-based and tritium-assisted fusion breeders. Nevertheless, we believe that our comparative analysis is sufficiently generic to provide reliable indications on the relative merits of the development of D-D-based and tritium-assisted FBs. These indications are summarized below, in addition to recommendations for future studies.

6.1. FUSION DEVICE AND PLASMA CHARACTERISTICS

1. The fusion power density and, even more so, the neutron source density offered by PCD plasmas can be significantly higher than those of the Cat-D plasmas subjected to the same confinement conditions. However, the PCD plasmas are characterized by a lower fusion energy gain.
2. Tritium assistance can significantly enhance the fusion energy gain, as well as the fusion power density and particularly the D-T fusion neutron yield of PCD plasmas.
3. The use of certain tritium-assisted SCD, or TCD modes of operation in the WILDCAT fusion power reactor promises enhanced power output, improved overall nuclear-to-electrical energy conversion efficiency, and, therefore, improved economics relative to the reference Cat-D WILDCAT.
4. Operating the WILDCAT design with PCD fuel cycles results in a lower first wall thermal loading--a design constraint--than in the reference Cat-D WILDCAT. The first wall neutron loading, however, is higher with the PCD modes of operation, but is still significantly lower than in the D-T STARFIRE tokamak design.

6.2. BLANKET AND FUEL PRODUCTION CHARACTERISTICS

1. The number of fissile atoms that can be produced per unit of nuclear energy released (i.e., $\overline{F}/\overline{W}$) is significantly higher for a D-D neutron than for a D-T neutron (even when the latter is not used to produce tritium).

2. The average $\bar{F}/\bar{W}$ attainable in D-D-based fusion breeders can be significantly higher than that attainable by D-T FBs.
3. Of the D-D-based fusion breeders, the highest fuel production-ability (as measured in terms of $\bar{F}/\bar{W}$, or support ratio) is offered by the SCD, while the lowest $\bar{F}/\bar{W}$ is offered by the Cat-D modes of operation.
4. The lower the blanket coverage efficiency, the larger the advantage of the D-D-based fusion drivers on the D-T fusion driver, in terms of the fuel production ability. For example, the support ratio of the SCD and TCD FBs is higher than that of a D-T FB by 80% and 50%, respectively, when the blanket coverage efficiency is 85%, and by close to 50% and 30% when the blanket completely surrounds the fusion neutron source.
5. In typical $ThO_2/Be/^3He$ or Li_2O fission-suppressed blankets considered here, the number of fissile atoms producible by a D-T neutron is approximately twice that producible by a D-D neutron.
6. The number of tritons that need be produced in beryllium fission-suppressed blankets to support a TCD mode of operation is typically 0.4 and 0.2 per D-T and D-D neutrons, respectively.
7. The optimal composition of the fission suppressed blankets considered in this study varies with the fusion fuel cycle. Thus, the ThO_2 volume fraction offering the highest support ratio is approximately 8%, 25% and 16% for D-T, Cat-D, and TCD fusion drivers, respectively.
8. The Th/Be atom density ratio offering the highest F/W for the D-T FB is about 50% higher than the corresponding ratio used for the reference blanket of the 1982 FB program.[5] If we had designed our blanket to have the Th/Be atom ratio of the reference FB program blanket, its $\bar{F}/\bar{W}$ would have been approximately 95% of the maximal attainable (see Fig. 8).
9. In blankets free of tritium production, the ThO_2 volume fraction offering the maximal fissile fuel production per given blanket energy is very closely the same ($\sim$25%) for the D-T and D-D neutrons.
10. The ThO_2 volume fraction offering the maximal F in tritium-production-free blankets is also approximately 25% for D-T neutrons, but the highest possible for D-D neutrons.

11. The neutron utilization in blankets that are designed to produce some tritium is superior to that in tritium-production-free blankets; the inclusion of even a relatively small amount of lithium or ^{3}He in the blanket can significantly reduce the parasitic neutron captures in the structural materials, as well as reduce the probability for thermal neutron fissions.
12. The use of enriched lithium is expected to improve the fuel production ability of blankets using lithium.
13. The support ratio offered by TCD FBs is somewhat higher than that offered by SCD-T FBs designed to have the same number of tritons produced per fusion neutron in the blanket. This is because of the lower binding energy released in neutron capture in ^{3}He (0.76 MeV) vs ^{6}Li (4.78 MeV). A unique potential merit of the SCD-T mode of operation, on the other hand, is that it can provide significant quantities of ^{3}He, possibly for D-^{3}He fusion power reactors.
14. It appears that a ^{3}He-to-T converter could be incorporated readily in the blanket/first wall system, leading to designs that are simpler and safer than those possible with lithium-containing blankets. The ^{3}He contained in the blanket also might be used for cooling both the blanket and the first wall.
15. It might be possible to confine the ^{3}He needed for the TCD mode of operation within a small volume near the first wall, or as part of the first-wall system. However, by distributing the ^{3}He throughout the blanket, it is possible to minimize the ^{3}He inventory. This inventory could be accumulated in a few months of operating the reactor in the SCD mode or possibly the SCD-T mode, (with the tritium supplied from external sources such as client fission reactors), or could be obtained from the decay of the tritium inventory of other fusion reactors.
16. It might be possible to produce part or all of the tritium needs of the FBs in the client fission reactors, without impairing their performance. We estimate that it might be possible to produce in LWRs operating in the Th/^{233}U fuel cycle as many as 0.38 tritons per fissile atom that they receive from the FB.

17. The tritium assistance offered by the client fission reactors can significantly improve the fuel production ability of the FBs and, possibly simplify the design of their blankets.
18. The symbiosis of LWRs with FBs operating with tritium assistance offers the best use of the neutrons, and possibly nuclear energy, generated in the LWRs and FBs.

6.3. ECONOMICS

1. Of the D-D-based fusion breeders considered, the best economic prospect is offered by tritium-assisted modes of operation, primarily the TCD. The economics of the Cat-D and SCD FBs is penalized primarily by their relatively low power density and, in the case of the Cat-D FB, by its relatively low support ratio as compared with the tritium-assisted PCD FBs.
2. The tritium-assisted PCD FBs considered appear profitable under the assumptions used with the market value of electricity and uranium expected in the mid-eighties.
3. The economics of the D-T, TCD, and other tritium assisted FBs based on the STARFIRE and WILDCAT designs, is significantly better than that of the reference STARFIRE and WILDCAT fusion power reactors.
4. The profitability of the TCD FBs is found to be lower by at least 25% and the levelized cost of fissile fuel is found to be higher by at least 25% than that of the D-T FB. The lower profitability of the tritium-assisted D-D-based FBs is due primarily to the relatively high capital cost of the WILDCAT design, almost 30% higher than that of the STARFIRE design. The higher energy recirculation requirements of the D-D-based FBs also penalize their economics.
5. It is quite conceivable that the profitability and levelized cost of fissile fuel for the tritium assisted D-D-based FBs could match or exceed that of the D-T FBs. Among the means expected to contribute to improvements in the relative economics of the TCD FBs are:
   (a) The design of a more compact and less expensive fusion driver.
   (b) Improved plasma confinement. This might be realized by higher β and the use of polarized plasmas among others.

(c) Use of tritium assistance from the client fission reactors.

(d) Sale of ^{3}He, in the case of SCD-T FBs.

(e) Increased uranium ore costs.

(f) Economy of scale effects on the FB fuel processing and fabrication costs. It ought to be noted, though, that the confinement of D-D-based plasmas is more demanding than the confinement of D-T plasmas; this might defer the commercialization of D-D FBs relative to that of the D-T FBs.

6. With their ~50% higher support ratio, the tritium assisted FBs make the fuel center concept for the development of the energy economy even more attractive. They enable us to reduce the total capacity and, possibly the number of sites required to provide the fuel needs of a given nuclear energy system, and thus possibly contribute to the practicality and acceptance of the fuel centers, and improve their safeguards. Therefore, if the economics of D-D-based and D-T FBs is found to be similar, the higher support ratio is a definite advantage.

7. With the ground rules and assumptions used in this work, both D-T and TCD FBs are found to be economically competitive with uranium enrichment facilities when the cost of U_3O_8 is \$40/lb. A detailed FB design study and a more thorough economic analysis is required, however, before firmer conclusions on the economics of FBs can be drawn.

6.4. RECOMMENDATIONS FOR FUTURE UNDERTAKINGS

The economic prospect and other merits of tritium-assisted D-D-based FBs (such as a high support ratio and ^{3}He source) are great enough to justify a thorough assessment of the promise of these FBs. For this assessment, it is essential that conceptual designs of tritium-assisted D-D-based FBs be undertaken, concentrating on the directions identified as most promising in this study. Primary emphasis should be given to the search for compact, relatively inexpensive fusion drivers, and for blanket designs that allow relatively low fuel processing and fabrication costs.

Among the specific future projects recommended are the following:

1. Study the economic prospects for tritium-assisted PCD FBs based on compact tokamaks and compact reversed-field pinch devices.
2. Search for promising realistic designs of blankets for tritium-assisted PCD FBs. Such designs should capitalize on the relieved tritium production requirements of these blankets and on the possibility of using ^{3}He instead of lithium. They should use fuel in a form that is relatively inexpensive to process and fabricate.
3. Identify promising and realistic ^{3}He-to-T converting systems, and evaluate their merits relative to Li-to-T converting systems. Particular emphasis needs to be given to low ^{3}He and tritium inventory converters. A significant problem is to determine how low the tritium inventory can be in a tritium-assisted D-D-based FB using a ^{3}He-to-T converter, relative to the tritium inventory in Cat-D and D-T FBs.
4. Investigate practical approaches for using otherwise wasted neutrons in client fission reactors (primarily LWRs) and study the effect of tritium assistance from client reactors on the properties and economics of D-D-based and D-T FBs.
6. Compare the economics of FBs and client fission reactors with the economics of fusion power reactors using the same fusion drivers.
5. Compare the economic prospects of the STARFIRE-based FB with that of tandem mirror-based FBs (of the type studied in Ref. 5) using common ground rules; the economics of the STARFIRE-based FB evaluated in this study appears better than that of the TMHR.[5] Our study, however, used simplified blanket and FB designs as well as a simplified economic model.
7. Determine reliably the value of fissile fuel for LWRs (i.e., the value the LWRs can afford to pay the FBs, when their COE is the same as the case in which the fissile fuel is provided by enrichment facilities). Consider LWRs operating both on the Th/^{233}U and the ^{238}U/Pu fuel cycles. Also, reevaluate the cost of the FB fuel processing and fabrication.
8. Assess the economic prospects of plutonium-producing D-D-based FBs. We anticipate that the fast-fission probability in ^{238}U-fueled blankets will be lower when driven by D-D-based fusion neutron sources.

9. For a similar reason, assess the promise of D-D-based FBs using fast-fission blankets.
10. Assess the economic value that can be assigned to the ^{3}He produced in the SCD (or SCD-T) FBs.

APPENDIX A. NUCLEAR DATA

This appendix supplements the information given in Section 3.2 concerning the preparation of the nuclear data used for this study.

A.1. ^{232}Th GAMMA PRODUCTION CROSS SECTIONS

The DLC-37 library[17] available for this study did not include gamma production cross sections for ^{232}Th. To calculate the energy released by the thorium in the form of gamma-rays, "gamma-energy release effective cross sections" for neutron capture and fission reactions (but not for inelastic scattering reactions) are defined as follows:

$$\sigma^g_{E_{n\gamma}} \equiv \int_{E_g^-}^{E_g^+} dE\, \sigma_{n\gamma}(E)\, E_{\gamma n}(E) \tag{A1}$$

and

$$\sigma^g_{E_{nf}} \equiv \int_{E_g^-}^{E_g^+} dE\, \sigma_{nf}(E)\, E_{\gamma f}(E) \quad . \tag{A2}$$

In the above, E_g^- and E_g^+ are the lower and upper boundaries of energy group g, whereas $E_{\gamma n}(E)$ and $E_{\gamma f}(E)$ are the gamma energy released from ^{232}Th caused by, respectively, the capture of and fission by a neutron of energy E. $E_{\gamma n}(E)$ is estimated using the expression[20]

$$E_{\gamma n}(E) \sim Q_{\gamma n} + E\,(1 - \frac{m}{M}) \sim 4.78 + 0.996\, E \text{ (MeV)}, \tag{A3}$$

where m and M are the neutron and reduced mass, respectively. $E_{\gamma f}(E)$ is significantly less sensitive to E--varying from 13.14 MeV for E = 14.0 MeV to 14.01 MeV for E = 3.35 MeV.[35] For the purpose of this study, we assumed $E_{\gamma f}$ to be a constant of 13.5 MeV.

It ought to be noted that whereas the contribution to the γ energy production by the entire blanket is accounted for using the γ-Kerma factor (calculating the gamma-energy deposited in the various system constituents), the contribution of ^{232}Th is accounted for by estimating the ^{232}Th gamma energy production assuming that most of it is being deposited in the system.

A.2. THORIUM FISSION SPECTRUM

The fission spectrum of ^{232}Th is calculated using a Maxwellian distribution and ENDF/B-IV data. A single spectrum corresponding to fissions induced by 14-MeV neutrons is used. This spectrum is given in Table A1.

Table A1. ^{232}Th fission spectrum.

Energy group[a]	χ (Normalized)	Energy group	χ (Normalized)
1	0.00010	7	0.04536
2	0.00024	8	0.33967
3	0.00050	9	0.26347
4	0.02771	10	0.05072
5	0.14556	11	0.00271
6	0.12396	12	0

[a]For energy group structure see Table 3.

A.3. AVERAGE THERMAL GROUP KERMA FACTORS AND ACTIVITIES

All MACKLIB-IV thermal neutron group constants including Kerma factors, tritium breeding, fission, and neutron capture cross sections are averaged over a Maxwellian spectrum at 800 K and corrected for cross-section deviations from a 1/v energy dependence; the correction factor is 0.5363g(800°K), where g is the Westcott factor.[36]

APPENDIX B. THE VALUE OF FISSILE FUEL

B.1. INTRODUCTION

The market price of electricity from modern LWRs in the mid eighties operating with the once-through fuel cycle is expected to be about 40 mills/kWh (see, for example, Refs. 5, 12, and 24). Estimated in this appendix is the price these LWRs could afford to pay for the FB-produced ^{233}U, so that the COE they produce will be 40 mills/kWh.

B.2. THE ONCE-THROUGH FUEL CYCLE

Consider conventional LWRs operating on the once-through fuel cycle. Table B1 summarizes the bus-bar generating cost of electricity production in LWRs to be commissioned in the mid- to late-eighties, along with the breakdown of the fuel cycle costs.[37]

Table B1. Estimated total bus-bar generating costs for future LWRs, in 1977 dollars.

Cost component	Remarks	Cost (mills/kWh)
Fuel Cycle Cost		
Yellow cake	at \$40 per pound	3.5
Conversion to UF_6	\$2.75 per pound	0.1
Enrichment	\$75 per SWU[a]	1.8
Fabrication	\$110 per kg HM[b]	0.7
Net salvage		1.0
Total		7.1
Carrying charges		26
Operation and maintenance		2
Bus-bar electricity cost		35

[a] Separative Work Unit (SWU).

[b] Heavy Metal (HM).

The fuel cycle cost components that could be saved if the fissile fuel is to be supplied by the FB include most of the yellow cake, conversion to UF_6, and enrichment. On the other hand, fabrication costs of ^{233}U bearing fuel is expected to be more expensive than that of conventional ^{235}U bearing fuel rods. The following is an attempt at cost adjustment:

- Amount of yellow cake necessary for the once-through denatured U fuel cycle is 18% of that required for the conventional ^{235}U-^{238}U fuel cycle. Hence, savings on yellow cake amounts to 2.86 mills/kWh.
- Fabrication cost of ^{233}U plus natural uranium fuel is assumed to be about 50% more expensive than that of conventional enriched uranium oxide fuel. This assumption is derived from Table II of Ref. 38 assigning $200/kg HM and $135/kg HM for the fabrication of PuO_2/UO_2 and UO_2 fuel rods, respectively. Consequently, we need to charge the ^{233}U-based fuel cycle cost for an extra of (200/135 - 1)*0.7 = 0.34 mills/kWh. Thus, the net credit that can be given to the ^{233}U fed by the FB is

Yellow cake		2.86
Conversion to UF_6		0.1
Enrichment		1.8
Fuel fabrication		-0.34
	TOTAL	4.42 mills/kWh

To determine the number of kWh that can be generated in the LWRs per gram of ^{233}U supplied by the FB, we assume (after Section IX.B.2.b of Ref. 5) the following:

- Fissile enrichment of 2.6%. Using natural uranium as a feed material for the once-through denatured uranium, the LWRs under consideration will require at least an extra 1.9% ^{233}U; in fact they may require a little more because ^{235}U is less effective than ^{233}U.
- Average discharge burnup of 33,000 MWD/T. Consequently, the energy value of ^{233}U is estimated to be $33{,}000*0.33*24*10^{-3}/0.019=$ $1.38*10^4$ kWh/g.

Thus, at a rate of 4.42 mills/kWh, the LWRs could afford to pay $4.42*10^{-3}*1.38*10^4=$ $60 per gram of ^{233}U.

B.3. DENATURED URANIUM FUEL CYCLE WITH RECYCLING

The total expenses associated with the recovery of the fissile fuel left in the LWR spent fuel are not expected to exceed the value of the fissile fuel recovered--say about $60 per gram. Otherwise there will be no economic incentive for fuel reprocessing. Thus, fuel recycling is expected to increase the price LWRs could pay to the FBs to above $60/g. The value of $75/g assumed for this study after Ref. 5 appears reasonable.

To determine the value of the fissile fuel bred in the FBs more reliably, the fuel cycle cost components need to be updated, and the economics of recycling in LWRs should be established and accounted for in detail.

APPENDIX C. PRODUCTION OF TRITIUM IN CLIENT REACTORS

Tables C1 and C2 give the fraction of fission neutrons lost by absorptions in boron and by leakage in the equilibrium and initial cycle of a typical PWR designed by Combustion Engineering, Inc.[32] These neutron fractions vary approximately linearly with burnup, so that the cycle average value can be well represented by the average of the Beginning of Cycle (BOC) and End of Cycle (EOC) values. Thus, by substituting ^{6}Li and/or ^{3}He for ^{10}B as the absorbing isotope, it is possible to utilize 3.6% of the neutrons of the equilibrium cycle for tritium production. If all the leaking neutrons could also be used for tritium production, this PWR could by-produce 7.5 tritons per 100 fission neutrons.

The tritium production ability in the initial core of the PWR is significantly higher (Table C2)--approximately 7.2 tritons per 100 fission neutrons. A fraction of the 3.89% neutrons leaking out of the initial core also might be used for tritium production.

In typical boiling water reactors (BWR) approximately 1% of the fission neutrons are absorbed, on the average, by the control rods.[32] An additional 4% of the neutrons are absorbed, on the average, by burnable poisons. Thus, close to 5% of the fission neutrons can be used for the production of tritium. Neutron leakage from BWRs of current design is smaller than the leakage from PWRs because of the use of natural uranium at the core periphery.

The substitution of lithium (or ^{3}He) for boron, gadolinium, or other elements used for absorption control, may not be straightforward in all cases. For example, it is not desirable to use lithium in the form of soluble poison because it will be too expensive to extract from the water the tritium produced, and possibly, too contaminating to leave high concentrations of tritium in the water. Nevertheless, we do see several approaches for the introduction of lithium into LWRs. Their description and evaluation is, however, beyond the scope of this work.

In conclusion, we expect that between 3.5% to 5% of the neutrons born in fission reactors can be used for tritium production in the LWRs control system. Because of the higher excess reactivity built into the initial core loading, additional tritium could be produced in the initial cycle of the LWRs. In addition, it might be possible to use a fraction of the leaking neutrons for tritium production. All the above is believed to be achievable without impairing the performance of the LWRs.

Table C1. Fraction of neutrons lost via absorptions in boron and via leakage in the equilibrium fuel cycle of a PWR.[a]

Parameter	Beginning of cycle	Cycle average	End of Cycle[b]
Absorptions for control			
Soluble boron	0.0712		0.0008
Cycle average		0.036	
Leakage			
Radial	0.0260		0.0283
Axial	0.0114		0.0122
Totals	0.0374		0.0405
Cycle average		0.039	
Total		0.1086	0.0413

[a]A Combustion Engineering, Inc. design for a 3800 MW_{th} PWR.[32]
[b]Cycle burnup is 10,267 MWD/T.

Table C2. Fraction of neutrons lost via absorptions in boron and via leakage in the initial core of a PWR.[a]

Parameter	Beginning of cycle	Cycle average	End of cycle[b]
Absorptions for Control			
Soluble boron	0.0570		0.0082
Boron shims	0.0734		0.0052
Total	0.1304		0.0134
Cycle average		0.072	
Leakage			
Radial	0.0353		0.0280
Axial	0.0038		0.0106
Total	0.0391		0.0386
Cycle average		0.0389	
Total 0.1695	0.0520		

[a]A Combustion Engineering, Inc. design for a 3800 MW_{th} PWR.[32]
[b]Cycle burnup is 17,300 MWD/T.

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